CHAPTER 5[†]: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STORM 100 System, including *the HI-STORM 100 overpack, HI-STORM 100S overpack, and* the *100-ton and 125-ton* HI-TRAC transfer cask*s*, is presented in this chapter. The HI-STORM 100 System is designed to accommodate different MPCs within ~~one~~ *two* ~~standard~~ HI-STORM overpack*s (the HI-STORM 100S overpack is a shorter version of the HI-STORM 100 overpack)*. The MPCs are designated as MPC-24, *MPC-24E and MPC-24EF* (24 PWR fuel assemblies), *MPC-32 (32 PWR fuel assemblies),* and MPC-68, *MPC-68F, and MPC-68FF* (68 BWR fuel assemblies). *The MPC-24E and MPC-24EF, which have a non-uniform internal pitch for improved criticality performance, are essentially identical to the MPC-24 from a shielding perspective. Therefore only the MPC-24 is analyzed in this chapter. Likewise, the MPC-68, MPC-68F and MPC-68FF are identical from a shielding perspective and therefore only the MPC-68 is analyzed. Throughout this chapter, unless stated otherwise, MPC-24 refers to either the MPC-24, MPC-24E, or MPC-24EF and MPC-68 refers to the MPC-68, MPC-68F, and MPC-68FF.*

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store ~~damaged~~ BWR *and PWR damaged* fuel assemblies and ~~BWR~~ fuel debris. Damaged fuel assemblies and fuel debris are defined in Section 2.1.3 and the ~~Technical Specifications of Chapter 12~~*approved contents section of Appendix B to the CoC*. Both damaged ~~BWR~~ fuel assemblies and ~~BWR~~ fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. DFCs containing *BWR* fuel debris must be stored in the MPC-68F *or MPC-68FF*. DFCs containing *BWR* damaged fuel assemblies may be stored in either the MPC-68, ~~or~~ the MPC-68F, *or the MPC-68FF. DFCs containing PWR fuel debris must be stored in the MPC-24EF while DFCs containing PWR damaged fuel assemblies may be stored in either the MPC-24E or MPC-24EF.* ~~Only the fuel assemblies in the Dresden 1 and Humboldt Bay fuel assembly classes identified in Table 2.1.2 are authorized as contents for storage in the HI-STORM 100 system as either damaged fuel or fuel debris.~~

The MPC-68, MPC-68F, and MPC-68FF are also capable of storing Dresden Unit 1 antimony-beryllium neutron sources and the single Thoria rod canister which contains 18 thoria rods that were irradiated in two separate fuel assemblies.

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter 1,* Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs) or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM 100 System has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices.

In order to offer the user more flexibility in fuel storage, the HI-STORM 100 System offers two different loading patterns in the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and the MPC-68FF. These patters are uniform and regionalized loading as described in Section 2.0.1 and 2.1.6. Since the different loading patterns have different allowable burnup and cooling times combinations, both loading patterns are discussed in this chapter.

The sections that follow will demonstrate that the design of the HI-STORM 100 dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The "controlled area" is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The cask vendor must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

3. Dose rates from the cask must be consistent with a well established "as low as reasonably achievable" (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than ~~5 Rem to the whole body or any organ~~*the limits specified in 10CFR 72.106.*

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.

This chapter contains the following information which demonstrates full compliance with the Standard Review Plan, NUREG-1536:

- A description of the shielding features of the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- The HI-STORM 100 System has been analyzed to show that the 10CFR72.104 and 10CFR72.106 controlled area boundary radiation dose limits are met during normal, off-normal, and accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters.
- Analyses are also presented which demonstrate that the storage of damaged fuel and fuel debris in the HI-STORM 100 System is ~~bounded by the BWR intact fuel analysis~~*acceptable* during normal, off-normal, and accident conditions.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains an analysis of the estimated dose at the controlled area boundary during normal, off-normal, and accident conditions from the release of radioactive materials. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM 100 System.

Chapter 10, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM 100 System are:

- Gamma radiation originating from the following sources

  1. Decay of radioactive fission products
  2. Secondary photons from neutron capture in fissile and non-fissile nuclides
  3. Hardware activation products generated during core operations

- Neutron radiation originating from the following sources

  1. Spontaneous fission
  2. α,n reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4. γ,n reactions (this source is negligible)
  5. *Dresden Unit 1 antimony-beryllium neutron sources*

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the steel structure of the MPC and the steel, lead, and water of the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water of the HI-TRAC transfer cask during loading, unloading, and transfer operations. Additionally, in the *125-ton* HI-TRAC top lid and transfer lid, a solid neutron shielding material, Holtite-A is used to thermalize the neutrons. Boron carbide, dispersed in the solid neutron shield material utilizes the high neutron absorption cross section of ^{10}B to absorb the thermalized neutrons.

The shielding analyses were performed with MCNP-4A [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 4.3 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis ~~intact~~ zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6~~, damaged,~~ and mixed oxide (MOX) fuel assemblies are the GE 6x6. *The GE 6x6 is also the design basis damaged fuel assembly for the Dresden Unit 1 and Humboldt Bay array classes.* Table 2.1.6 specifies the acceptable intact zircaloy clad fuel characteristics for storage. Table 2.1.7 specifies the acceptable damaged ~~and MOX zircaloy clad~~ fuel characteristics for storage.

The design basis ~~intact~~ stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. Table 2.1.8 specifies the acceptable fuel characteristics of stainless steel clad fuel for storage.

The MPC-24, *MPC-24E, MPC-24EF, MPC-32, MPC-68,* and MPC-68*FF* are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. *The approved contents section of Appendix B to the CoC* ~~Figure 2.1.6~~ specifies the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in the*se MPCs*~~MPC-24 and the MPC-68~~. ~~Table 2.1.8~~*Appendix B to the CoC also* specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. The values in ~~Figure 2.1.6 and the Table 2.1.8~~*Appendix B to the CoC* were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC.

The dose rates surrounding the HI-STORM overpack are very low, and thus, the shielding analysis of the HI-STORM overpack conservatively considered the burnup and cooling time combinations listed below, which bound the acceptable burnup levels and cooling times from ~~Figure 2.1.6 and Table 2.1.8~~*Appendix B to the CoC.* This large conservatism is included in the analysis of the HI-STORM overpack to unequivocally demonstrate that the HI-STORM overpack meets the Part 72 dose requirements.

Zircaloy Clad Fuel		
MPC-24	***MPC-32***	**MPC-68**
~~45,000~~*52,500* MWD/MTU 5 year cooling	*45,000 MWD/MTU 5 year cooling*	~~45,000~~*47,500* MWD/MTU 5 year cooling
Stainless Steel Clad Fuel		
MPC-24	***MPC-32***	**MPC-68**
40,000 MWD/MTU 8 year cooling	*40,000 MWD/MTU 9 year cooling*	22,500 MWD/MTU 10 year cooling

~~Technical Specification Table 2.1-4, Appendix 12.A, requires that, in the MPC-24, for a minimum cooling time of 5 years, the maximum burnup is 31,300 MWD/MTU, and for 15-year cooling the maximum burnup is 44,700 MWD/MTU.~~ *The burnup and cooling time combinations analyzed for zircaloy clad fuel produce dose rates at the midplane of the HI-STORM overpack which bound all uniform and regionalized loading burnup and cooling time combinations listed in Appendix B to the CoC.* ~~The HI-STORM shielding analysis was performed for the absolute maximum allowable burnup (i.e., 45,000 MWD/MTU) and absolute minimum cooling time (i.e., 5 years), the combination of which conservatively bounds the allowable burnup and cooling time~~

~~combinationse.~~ Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, *MPC-32, and MPC-68.*

~~Technical Specification Table 2.1-4, Appendix 12.A, requires that, in the MPC-68, for a minimum cooling time of 5 years, the maximum burnup is 29,900 MWD/MTU, and for 15 year cooling the maximum burnup is 41,700 MWD/MTU. The HI-STORM shielding analysis was performed for the absolute maximum allowable burnup (i.e., 45,000 MWD/MTU) and absolute minimum cooling time (i.e., 5 years), the combination of which conservatively bounds the allowable burnup and cooling time combinations. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-68.~~

The dose rates surrounding the HI-TRAC transfer cask are significantly higher than the dose rates surrounding the HI-STORM overpack, and although no specific regulatory limits are defined, dose rates are based on the ALARA principle. Therefore, the cited dose rates were based on the actual burnups and cooling times requested in ~~the Technical Specifications~~*Appendix B to the CoC*. Two different burnup and cooling times, *listed below,* were analyzed for the MPC-24, *MPC-32,* and the MPC-68 *in the 100-ton HI-TRAC. The burnups corresponding to 5-year cooling times produce dose rates at 1 meter from the surface of the overpack, for the locations reported in this chapter, which bound the dose rates from all other uniform loading burnup and cooling time combinations listed in Appendix B to the CoC. Since it is reasonable to assume that the majority of fuel which will be loaded in casks will be 10 years or older, the dose rates from conservative burnups for 10-year cooling are also presented in this chapter.* ~~These burnup and cooling times, listed below, are for the minimum and maximum allowable burnups as shown in Figure 2.1.6. Conservatively, the maximum allowable burnup was analyzed with a relatively short cooling time (9 and 12 years as opposed to 15 years for the MPC-24 and MPC-68 respectively).~~

100-ton HI-TRAC		
MPC-24	***MPC-32***	**MPC-68**
~~35,000~~*42,500* MWD/MTU 5 year cooling	*32,500 MWD/MTU* *5 year cooling*	~~30,000~~*40,000* MWD/MTU 5 year cooling
~~45,000~~*52,500* MWD/MTU ~~9~~*10* year cooling	*45,000 MWD/MTU* *10 year cooling*	*50,000*~~45,000~~ MWD/MTU ~~12~~ *10* year cooling

The *100-ton* HI-TRAC with the MPC-24 has higher dose rates at the mid-plane than the *100-ton* HI-TRAC with the *MPC-32 or the* MPC-68. Therefore, the MPC-24 results *for 5-year cooling* are presented in this section and the MPC-24 was used for the dose exposure estimates in Chapter 10. The *MPC-32 results,* MPC-68 results, *and additional MPC-24 results* are provided in Section 5.4 for comparison.

The 100-ton HI-TRAC dose rates bound the 125-ton HI-TRAC dose rates for the same burnup and cooling time combinations. Therefore, for illustrative purposes, the MPC-24 was the only MPC analyzed in the 125-ton HI-TRAC. Dose rates are presented for two different burnup and cooling time combinations for the MPC-24 in the 125-ton HI-TRAC: 42,500 MWD/MTU with 5-year cooling and 57,500 MWD/MTU with 12-year cooling. The dose rates for the later combination are presented in this section because it produces the highest dose rate at the cask midplane. Dose rates for the other burnup and cooling time combination are presented in Section 5.4. ~~The burnup and cooling times which produce the highest dose rates on the side of the HI-TRAC are presented in this section for the 100-ton and 125-ton HI-TRAC. Dose rates for the additional burnup and cooling times are presented in Section 5.4.~~

As a general statement, the dose rates for uniform loading presented in this chapter bound the dose rates for regionalized loading at 1 meter distance from the overpack. Therefore, dose rates for specific burnup and cooling time combinations in a regionalized loading pattern are not presented in this chapter. Section 5.4.9 provides an additional brief discussion on regionalized loading.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STORM 100 System. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other *critical* organ.

2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 10 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM 100 System in Section 2.3.5.2 as: ~~40~~ *60* mrem/hour on the radial surface of the overpack, 60 mrem/hour at the openings of the air vents, and ~~10~~ *60* mrem/hour on the top of the overpack.

The HI-STORM overpack dose rates presented in this section are conservatively ~~calculated at 45,000 MWD/MTU and 5 year cooling~~*evaluated* for ~~both~~ *the MPC-32,* the MPC-68, and the MPC-24. All burnup and cooling time combinations analyzed bound the allowable burnup and cooling times specified in ~~Chapters 2 and 12~~*Appendix B to the CoC.*

Figure 5.1.1 *and 5.1.12* identif*y*~~ies~~ the locations of the dose points referenced in the dose rate summary tables for the HI-STORM *100 and HI-STORM 100S* overpack*s, respectively*. Dose Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the peak dose location above the overpack shield block. For the adjacent top dose, this dose point is located over the air annulus between the MPC and the overpack. *Dose Point #4a in Figure 5.1.12 is located directly above the exit duct and next to the concrete shield block.* The dose values reported at the locations shown on Figure 5.1.1 *and 5.1.12* are averaged over a region that is approximately 1 foot in width.

The total dose rates presented in this chapter for the MPC-24 and MPC-32 are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Section 5.2.4.1. This is conservative because it is not expected that the cooling times for both the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values.

Tables 5.1.*1*~~2~~ and 5.1.3 provide the maximum dose rates adjacent to the *HI-STORM 100S* overpack during normal conditions for ~~each of the MPCs~~*the MPC-32 and MPC-68.* Tables 5.1.~~5~~*4* and 5.1.6 provide the maximum dose rates at one meter from ~~the overpack~~*the HI-STORM 100S overpack. Tables 5.1.2 and 5.1.5 provide the maximum dose rates adjacent to and one meter from the HI-STORM 100 overpack for the MPC-24.*

Although the dose rates for the MPC-~~68~~*32 in HI-STORM 100s* are ~~slightly greater~~*equivalent to or greater* than those for the MPC-24 *in HI-STORM 100*~~(for identical burnup and cooling time)~~, as shown in Tables 5.1.~~2~~*1*, 5.1.~~3~~*2*, 5.1.~~5~~*4*, and 5.1.~~6~~*5*, the MPC-24 was used in the calculations for the dose rates at the controlled area boundary. The MPC-24 was chosen because, for a given cooling time, the MPC-24 has a higher allowable burnup than the *MPC-32 or the* MPC-68 (see ~~Table 2.1-4 of Appendix 12A~~*Appendix B to the CoC*). Consequently, for the allowable burnup and cooling times, the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 *and MPC-32. The dose rates at the controlled area boundary were calculated for the HI-STORM 100 overpack rather than the HI-STORM 100S overpack. The difference in height will have little impact on the dose rates at the controlled area boundary since the surface dose rates are very similar. The controlled area boundary dose rates were also calculated without including non-fuel hardware. This is acceptable because the dose rates for the HI-STORM 100 overpack calculated in Table 5.1.2 without BPRAs are conservative enough to* *bound the dose rates for actual burnup and cooling times from Appendix B to the CoC including BPRAs.*

Table 5.1.7 provides dose rates adjacent to *and one meter from the* ~~both the 125-ton and~~ 100-ton HI-TRAC. Table 5.1.8 provides dose rates *adjacent to and* one meter from ~~both~~ the 125-ton ~~and 100-ton~~ HI-TRAC. Figures 5.1.2 and 5.1.4 identify the locations of the dose points referenced in Tables 5.1.7 and 5.1.8 for the HI-TRAC 125-ton and 100-ton transfer casks, respectively. The dose rates listed in Tables 5.1.7 and 5.1.8 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. The dose rates below the HI-TRAC (Dose Point #5) are provided for two conditions. The first condition is when the pool lid is in use and the second condition is when the transfer lid is in use. The calculational model of the 100-ton HI-TRAC included a concrete floor positioned 6 inches (the typical carry height) below the pool lid to account for ground scatter. As a result of the modeling, the dose rate at 1 meter from the pool lid for the 100-ton HI-TRAC was not calculated. The dose rates provided in Tables 5.1.7 and 5.1.8 are for the MPC-24 with design basis fuel at burnups and cooling times, based on the allowed burnup and cooling times specified in ~~the Technical Specifications~~ *Appendix B to the CoC*, that result in dose rates that are generally higher in each of the two HI-TRAC designs. The burnup and cooling time combination used for both the 100-ton and 125-ton HI-TRAC was chosen based on ~~Figure 2.1.6~~ *the allowable burnup and cooling times in Appendix B to the CoC.* Results for other burnup and cooling times and for the MPC-68 *and MPC-32* are provided in Section 5.4.

Because the dose rates for the 100-ton HI-TRAC transfer cask are significantly higher than the dose rates for the 125-ton HI-TRAC or the HI-STORM overpack, it is important to understand the behavior of the dose rates surrounding the external surface. To assist in this understanding, several figures, showing the dose rate profiles on the top, bottom and sides of the 100-ton HI-TRAC transfer cask, are presented below. *The figures discussed below were all calculated without the gamma source from BPRAs.*

Figure 5.1.5 shows the dose rate profile at 1 foot from the side of the 100-ton HI-TRAC transfer cask with the MPC-24 for 35,000 MWD/MTU and 5 year cooling. This figure clearly shows the behavior of the total dose rate and each of the dose components as a function of the cask height. To capture the effect of scattering off the concrete floor, the calculational model simulates the 100-ton HI-TRAC at a height of 6 inches (the typical cask carry height) above the concrete floor. As expected, the total dose rate on the side near the top and bottom is dominated by the Co-60 gamma dose component, while the center dose rate is dominated by the fuel gamma dose component.

The total dose rate and individual dose rate components on the surface of the pool lid on the 100-ton HI-TRAC are provided in Figure 5.1.6, illustrating the significant reduction in dose rate with increasing distance from the center of the pool lid. Specifically, the total dose rate is shown to drop by a factor of more than 20 from the center of the pool lid to the outer edge of the HI-TRAC. Therefore, even though the dose rate in Table 5.1.7 at the center of the pool lid is substantial~~(3.5 Rem)~~, the dose rate contribution, from the pool lid, to the personnel exposure is minimal.

The behavior of the dose rate 1-foot from the transfer lid is shown in Figure 5.1.7. Similarly, the total dose rate and the individual dose rate components 1-foot from the top lid, as a function of distance from the axis of the 100-ton HI-TRAC, are shown in Figure 5.1.8. For both lids (transfer and top), the reduction in dose rate with increased distance from the cask axial centerline is substantial.

To reduce the dose rate above the water jacket, a localized temporary shield ring, described in Chapter 8, may be employed on the 125-ton HI-TRAC and ~~will be mandatory~~ on the 100-ton HI-TRAC. This temporary shielding, which is water, essentially extends the water jacket to the top of the HI-TRAC. The effect of the temporary shielding on the side dose rate above the water jacket (in the area around the lifting trunnions and the upper flange) is shown on Figure 5.1.9, which shows the dose profile on the side of the 100-ton HI-TRAC with the temporary shielding installed. For comparison, the total dose rate without temporary shielding installed is also shown on Figure 5.1.9. The results indicate that the temporary shielding reduces the dose rate by approximately a factor of 2 in the area above the water jacket.

To illustrate the reduction in dose rate with distance from the side of the 100-ton HI-TRAC, Figure 5.1.10 shows the total dose rate on the surface and at distances of 1-foot and 1-meter.

Figure 5.1.11 plots the total dose rate at various distances from the bottom of the transfer lid, including distances of 1, 5, 10, and 15 feet. Near the transfer lid, the total dose rate is shown to decrease significantly as a function of distance from the 100-ton HI-TRAC axial centerline. Near the axis of the HI-TRAC, the reduction in dose rate from the 1-foot distance to the 15-foot distance is approximately a factor of 15. The dose rate beyond the radial edge of the HI-TRAC is also shown to be relatively low at all distances from the HI-TRAC transfer lid. Thus, prudent transfer operating procedures will employ the use of distance to reduce personnel exposure. In addition, when the HI-TRAC is in the horizontal position and is being transported on site, a missile shield ~~(large steel plate) will~~*may* be positioned in front of the HI-TRAC transfer lid. ~~This missile shield will~~*If present, this shield would* also serve as temporary gamma shielding which ~~will~~ *would* greatly reduce the dose rate in the vicinity of the transfer lid. For example, if the missile shield was a 2 inch thick steel plate, the gamma dose rate would be reduced by approximately 90%.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. As mentioned, only the MPC-24 was used in the calculation of the dose rates at the controlled area boundary. Table 5.1.9 presents the annual dose to an individual from a single HI-STORM cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. These values were conservatively calculated for a burnup of ~~45,000~~*52,500* MWD/MTU and a 5-

year cooling time. In addition, the annual dose was calculated for a burnup of 45,000 MWD/MTU and a 9 year cooling time. *BPRAs were not included in these dose estimates.* It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212, as stated in Chapter 12, "Operating Controls and Limits". The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the cask array configurations provided in Table 5.1.9. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Section 5.2 lists the gamma and neutron sources for the design basis ~~intact and damaged~~ fuels. Since the source strengths of the *GE 6x6 intact and* damaged fuel and the *GE 6x6* MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the ~~damaged and MOX fuel~~ dose rates *from the GE 6x6 fuels* for normal conditions are bounded by the MPC-68 analysis with the design basis intact fuel. Therefore, no explicit analysis of the MPC-68~~F~~ with either *GE 6x6 intact or* damaged or *GE 6x6* MOX fuel for normal conditions is required to demonstrate that the MPC-68 with *GE 6x6* ~~damaged or MOX~~ fuels will meet the normal condition regulatory requirements. *Section 5.4.2 evaluates the effect of generic damaged fuel in the MPC-24E and the MPC-68.*

Section 5.2.6 lists the gamma and neutron sources from the Dresden Unit 1 Thoria rod canister and demonstrates that the Thoria rod canister is bounded by the design basis Dresden Unit 1 6x6 intact fuel.

Section 5.2.4 presents the Co-60 sources from the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM 100 System. Section 5.4.6 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

Section 5.4.7 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Section 5.2.3 lists the gamma and neutron sources for the design basis ~~intact~~ stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.4.

The analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.104 limits and ALARA practices.

5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

> Any individual located on or beyond the nearest boundary of the controlled area ~~shall~~ *may* not receive *from any design basis accident the more limiting of a total effective dose equivalent of 5 Rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem. The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem.* ~~a dose greater than 5 Rem to the whole body or any organ from any design basis accident.~~ The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled are shall be at least 100 meters.

Design basis accidents which may affect the HI-STORM overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary, adjacent, and one meter doses for the loaded HI-STORM overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the fuel spacers. The HI-STAR 100 System (Docket Number 72-1008) documentation provides analysis to demonstrate that the fuel spacers will not fail under any normal, off-normal, or accident condition of storage. Chapter 3 also shows that the HI-TRAC inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell could be experienced. However, the localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. *The burnup and cooling time combinations used in Table 5.1.10 were the combinations that resulted in the highest post-*

accident condition dose rates. These burnup and cooling time combinations do not necessarily correspond to the burnup and cooling time combinations that result in the highest dose rate during normal conditions. Scaling this accident dose rate by the dose rate reduction seen in HI-STORM yields a dose rate at the 100 meter controlled area boundary that would be approximately 0.~~8~~*1.47*† mrem/hr for the HI-TRAC accident condition. At this dose rate, it would take ~~6250~~ *3401* hours (~~~260~~ *141* days) for the dose at the controlled area boundary to reach 5 Rem. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem at the controlled area boundary for the short duration of the accident.

The consequences of the design basis accident conditions for the MPC-68 *and MPC-24E* storing damaged fuel and the MPC-68F*, MPC-68FF, or MPC-24EF* storing damaged fuel and/or fuel debris differ slightly from those with intact fuel. It is conservatively assumed that during a drop accident (vertical, horizontal, or tip-over) the damaged fuel collapses and the pellets rest in the bottom of the damaged fuel container. ~~Since the damaged and MOX fuels are both Dresden 1 fuel, the MOX fuel can also be considered damaged fuel.~~ Analyses in Section 5.4.2 demonstrates that the damaged fuel in the post-accident condition ~~has lower source terms (both gamma and neutron) per inch than the intact BWR design basis fuel~~ *does not significantly affect the dose rates around the cask.* Therefore, the damaged fuel post-accident dose rates are bounded by the ~~BWR~~ intact fuel post-accident dose rates.

Analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.106 limits.

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† ~~1089~~ *2083.22* mrem/hr (Table 5.1.10) x [~~109~~ *129* mrem/yr (Table 5.4.7) / 8760 hrs / ~~17.4~~*20.9* mrem/hr (Table 5.1.5)]

Table 5.1.1

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK
FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point*† *Location	***Fuel Gammas*†† *(mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***	***Totals with BPRAs (mrem/hr)***
1	*10.45*	*16.44*	*7.18*	*34.07*	*34.94*
2	*37.18*†††	*0.05*	*2.13*	*39.36*	*45.14*
3	*11.49*	*17.00*	*5.65*	*34.14*	*41.78*
4	*2.42*	*1.02*	*2.12*	*5.56*	*6.23*
4a	*3.93*	*9.58*	*29.15*	*42.66*	*47.26*

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† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for 8.~~7~~*5* % of this dose rate.

Table 5.1.2

DOSE RATES ADJACENT TO *HI-STORM 100* OVERPACK
FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~45,000~~*52,500* MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	***Totals with BPRAs (mrem/hr)***
1	*7.20*	*5.34*	*4.46*	*17.00*	*17.35*
2	*37.65*†††	*0.03*	*3.04*	*40.72*	*45.77*
3	*4.87*	*3.52*	*2.23*	*10.61*	*12.16*
4	*1.28*	*0.39*	*5.82*	*7.49*	*7.70*

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for ~~8.6~~*8.0* % of this dose rate.

Table 5.1.3

DOSE RATES ADJACENT TO *HI-STORM 100S* OVERPACK FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~45,000~~*47,500* MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*10.45*	*12.45*	*9.57*	*32.46*
2	*33.87*	*0.01*	*2.90*	*36.78*
3	*4.74*	*16.03*	*3.98*	*24.75*
4	*1.44*	*1.19*	*1.63*	*4.26*
4a	*1.14*	*9.76*	*20.04*	*30.94*

† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.4

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK FOR NORMAL CONDITIONS MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING BURNUP AND COOLING TIME 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	***Fuel Gammas†† (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***	***Totals with BPRAs (mrem/hr)***
1	*5.66*	*5.51*	*1.03*	*12.20*	*12.96*
2	*18.83†††*	*0.66*	*0.90*	*20.39*	*23.47*
3	*4.95*	*4.85*	*0.84*	*10.65*	*13.20*
4	*0.77*	*0.29*	*1.02*	*2.08*	*2.23*

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† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for 8.6 % of this dose rate.

Table 5.1.5

DOSE RATES AT ONE METER *FROM HI-STORM 100 OVERPACK*
FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~45,000~~*52,500* MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	***Totals with BPRAs (mrem/hr)***
1	*5.73*	*3.18*	*0.87*	*9.79*	*10.29*
2	*19.38*†††	*0.27*	*1.26*	*20.90*	*23.48*
3	*3.28*	*2.29*	*0.34*	*5.91*	*7.05*
4	*0.58*	*0.18*	*1.77*	*2.53*	*2.63*

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

††† The cobalt activation of incore grid spacers accounts for ~~8.5~~*8.0* % of this dose rate.

Table 5.1.6

DOSE RATES AT ONE METER *FROM HI-STORM 100S OVERPACK*
FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~45,000~~*47,500* MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*5.69*	*3.92*	*1.27*	*10.89*
2	*16.77*	*0.29*	*1.22*	*18.27*
3	*2.57*	*5.07*	*0.53*	*8.17*
4	*0.43*	*0.29*	*0.81*	*1.53*

† Refer to Figure 5.1.12.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.7

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
42,500 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	37.30	12.83	626.44	174.37	850.94	858.92
2	985.63†	53.56	0.92	89.47	1129.58	1355.26
3	15.72	2.01	546.21	217.69	781.64	998.97
3 (temp)	6.66	4.45	190.43	2.22	203.75	279.72
4	24.53	0.99	278.26	183.08	486.86	602.46
4 (outer)	7.01	0.62	69.28	123.77	200.68	229.77
5 (pool lid)	174.22	17.77	3159.24	1210.89	4562.13	4622.29
5 (transfer)	293.92	0.64	3527.96	676.12	4498.64	4561.70
5(t-outer)	77.59	0.34	378.49	269.68	726.10	742.20
ONE METER FROM THE 100-TON HI-TRAC						
1	128.52	7.24	91.31	28.36	255.43	284.54
2	427.92†	16.63	7.07	32.83	484.45	582.43
3	54.55	3.93	83.00	14.03	155.51	203.26
3 (temp)	54.25	4.16	69.21	4.88	132.50	174.84
4	8.40	0.17	85.91	45.49	139.97	175.78
5 (transfer)	122.33	0.14	1502.34	188.57	1813.38	1837.76
5(t-outer)	19.43	0.62	133.57	54.20	207.82	210.72

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 3(temp) represents dose location 3 with temporary shielding installed.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

† *The cobalt activation of incore grid spacers accounts for 12.3% of the surface and one-meter dose rates.*

Table 5.1.8

DOSE RATES FROM THE 125-TON HI-TRAC FOR NORMAL CONDITIONS MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL 57,500 MWD/MTU AND 12-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 125-TON HI-TRAC						
1	2.26	25.55	48.08	150.18	226.07	227.20
2	22.55†	72.90	0.00	89.90	185.36	199.50
3	0.25	3.05	17.34	249.62	270.25	284.99
4	8.37	3.12	118.06	285.41	414.96	522.13
4 (outer)	0.94	2.23	14.66	5.98	23.82	37.00
5 (pool)	10.77	1.36	157.53	1070.88	1240.55	1247.70
5 (transfer)	8.19	1.54	144.40	158.05	312.20	316.43
ONE METER FROM THE 125-TON HI-TRAC						
1	3.00	9.26	5.26	23.93	41.44	43.28
2	9.73†	22.83	0.17	32.42	65.15	71.27
3	1.14	5.46	3.47	21.80	31.87	35.70
4	2.35	0.75	28.42	29.32	60.85	86.53
5 (transfer)	4.78	0.29	73.03	28.25	106.35	109.10

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

† *The cobalt activation of incore grid spacers accounts for 15.5% of the surface and one-meter dose rates.*

Table 5.1.9

DOSE RATES FOR ARRAYS OF MPC-24
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES

Array Configuration	1 cask	2x2	2x3	2x4	2x5
***52,500* MWD/MTU AND 5-YEAR COOLING**					
Annual Dose (mrem/year)†	*20.19*	*23.83*	*19.13*	*14.91*	*18.64*
Distance to Controlled Area Boundary (meters)††,†††	200	250	300	*350*	350
45,000 MWD/MTU AND 9-YEAR COOLING					
Annual Dose (mrem/year) †	16.03	16.95	12.19	16.26	20.32
Distance to Controlled Area Boundary (meters) ††	150	200	250	250	250

† 8760 hr. annual occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 5-year cooling, as specified in the ~~Technical Specifications~~*Appendix B to the CoC*, is lower than the burnup used for this analysis.

Table 5.1.10

DOSE RATES AT ONE METER FROM HI-TRAC
FOR ACCIDENT CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
AT BOUNDING BURNUP AND COOLING TIMES

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	***Totals with BPRAs (mrem/hr)***
125-TON HI-TRAC					
***57,500* MWD/MTU AND *12*-YEAR COOLING**					
2 (Accident Condition)	*19.60*	*0.43*	*1284.98*	*1305.01*	*1317.24*
2 (Normal Condition)	*32.56*	*0.17*	*32.42*	*65.15*	*71.27*
100-TON HI-TRAC					
***57,500* MWD/MTU AND *12*-YEAR COOLING**					
2 (Accident Condition)	*280.91*	*5.89*	*1621.02*	*1907.82*	*2083.22*
2 (Normal Condition)	*182.83*	*4.76*	*59.80*	*247.39*	*345.37*

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

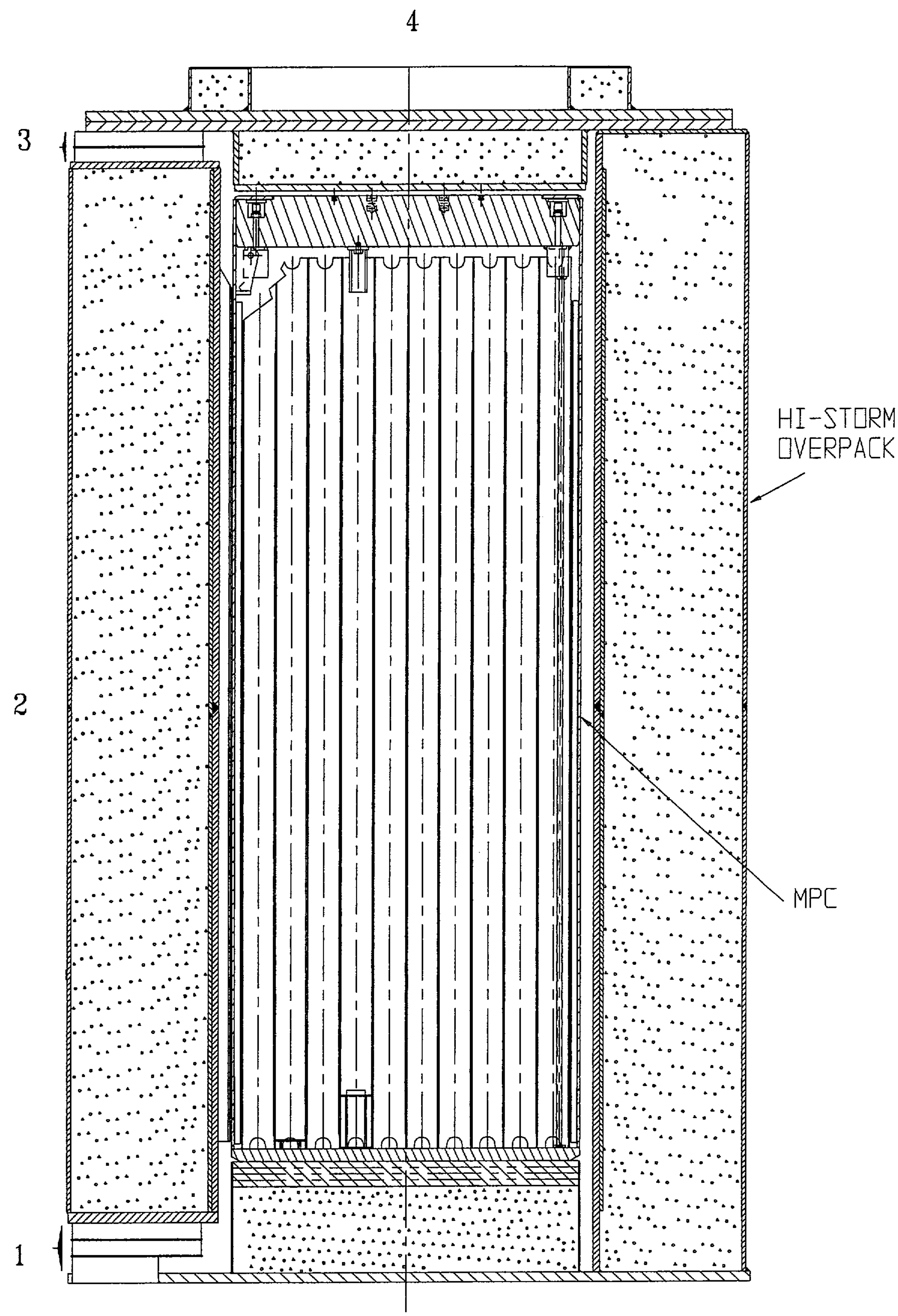

FIGURE 5.1.1; CROSS SECTION ELEVATION VIEW OF HI-STORM 100 OVERPACK WITH DOSE POINT LOCATION

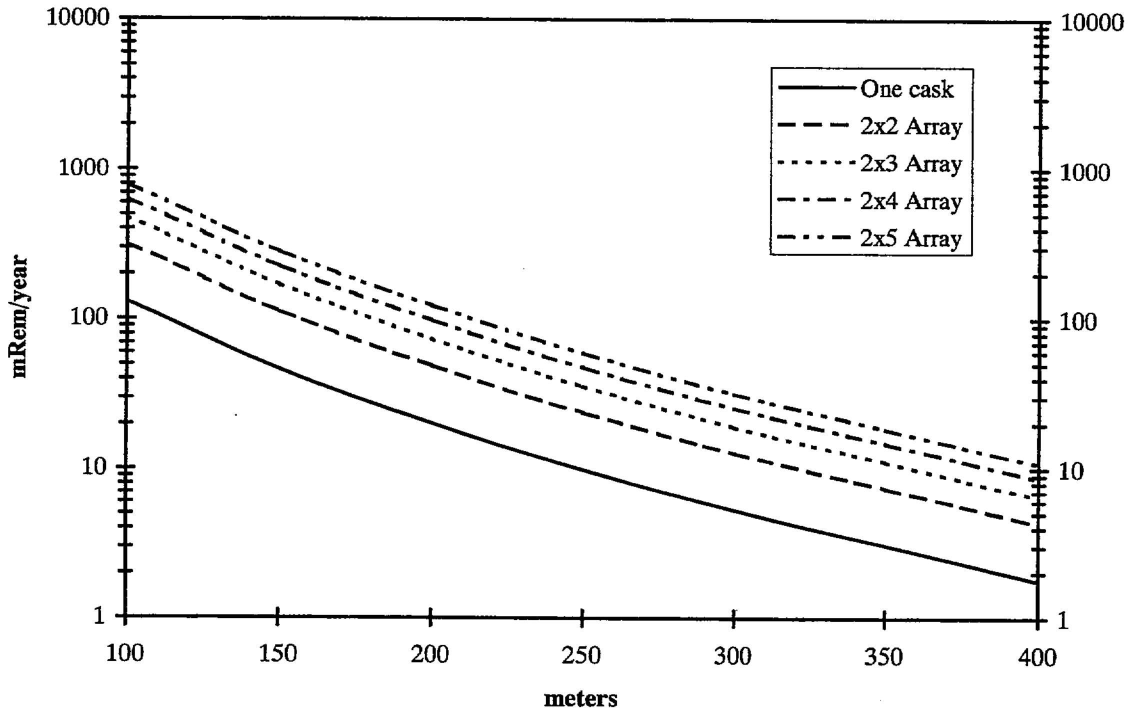

FIGURE 5.1.3; ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-24 FOR 52,500 MWD/MTU AND 5-YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)

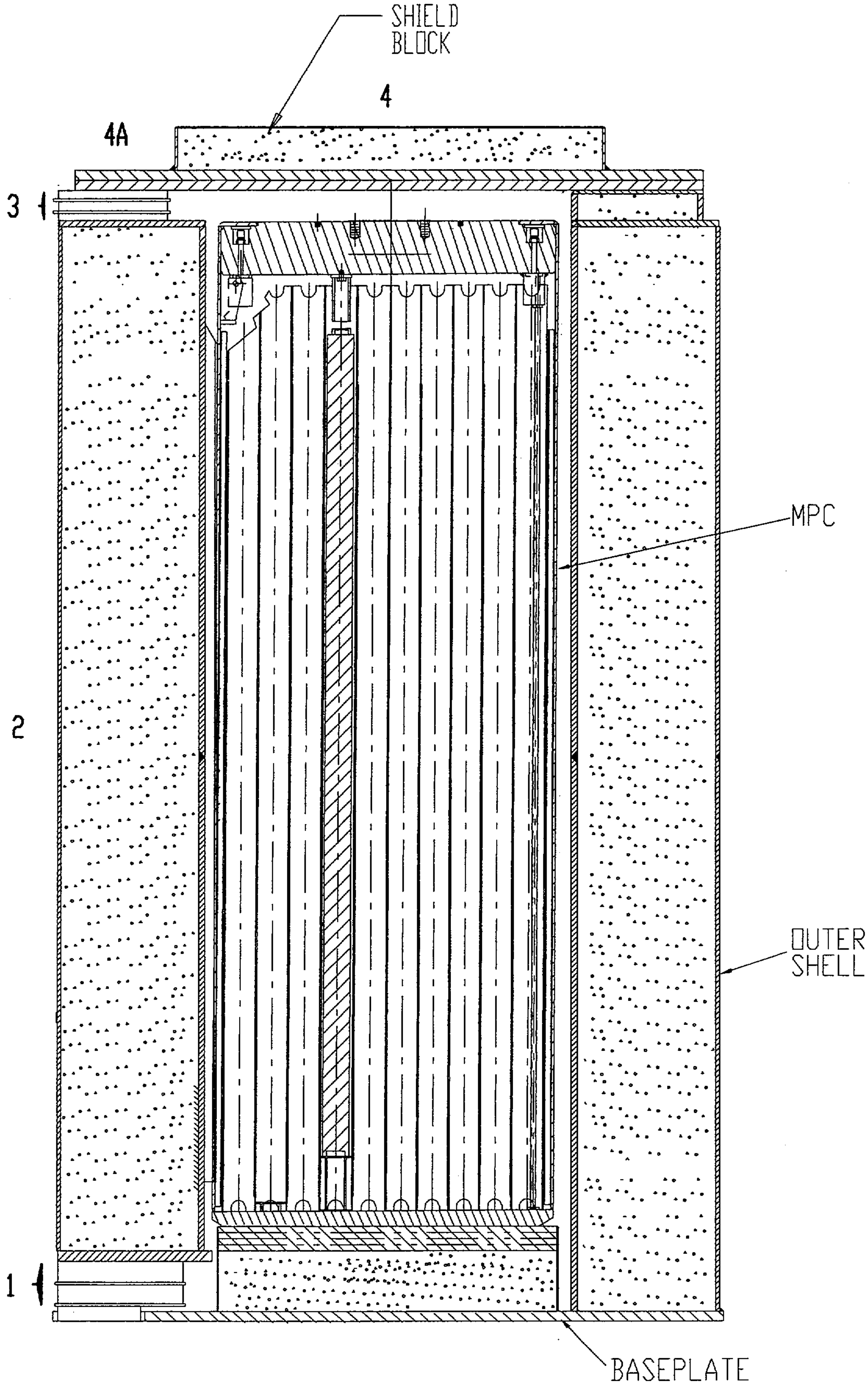

FIGURE 5.1.12j CROSS SECTION ELEVATION VIEW OF THE HI-STORM 100S OVERPACK WITH DOSE POINT LOCATION

5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system [5.1.2, 5.1.3]. *SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.*

Sample input files for SAS2H and ORIGEN-S are provided in Appendices 5.A and 5.B, respectively. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from ^{60}Co activity of the steel structural material in the fuel element above and below the active fuel region. The third source is from (n,γ) reactions described below.

A description of the design basis ~~intact~~ zircaloy clad fuel for the source term calculations is provided in Table 5.2.1. The PWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load from the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun. The BWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load from the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8. Multiple SAS2H and ORIGEN-S calculations were performed to confirm that the B&W 15x15 and the GE 7x7, which have the highest UO_2 mass, bound all other PWR and BWR fuel assemblies, respectively. Section 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

The design basis Humboldt Bay and Dresden 1 6x6 fuel assembly~~, which is also the design basis damaged fuel assembly for the Humboldt Bay and Dresden 1 damaged fuel or fuel debris,~~ is described in Table 5.2.2. ~~The design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.~~ The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.21 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6~~, damaged,~~ and MOX fuel assemblies which are smaller than the GE 7x7, are assumed to have the same hardware characteristics as the GE 7x7. This is conservative because the larger hardware mass of the GE 7x7 results in a larger ^{60}Co activity.

The design basis stainless steel clad fuel assembly for the *Indian Point 1,* Haddam Neck and San Onofre 1 assembly classes is described in Table 5.2.3. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.

The design basis assemblies mentioned above are the design basis assemblies for both intact and damaged fuel and fuel debris for their respective array classes. Analyses of damaged fuel is presented in Section 5.4.2.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1, 5.2.2, 5.2.3, and 5.2.21 resulted in conservative source term calculations.

Sections 5.2.1 and 5.2.2 describe the calculation of gamma and neutron source terms for zircaloy clad fuel while Section 5.2.3 discusses the calculation of the gamma and neutron source terms for the stainless steel clad fuel.

5.2.1 Gamma Source

Tables 5.2.~~4~~5 ~~and~~ *through* 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis ~~intact~~ zircaloy clad fuels at varying burnups and cooling times. Tables 5.2.7 and 5.2.22 provides the gamma source in MeV/s and photons/s for the design basis ~~damaged~~ *6x6* and MOX fuel, respectively.

Specific analysis for the HI-STORM 100 System, which includes the HI-STORM storage overpack*s* and the HI-TRAC transfer cask*s*, was performed to determine the dose contribution from gammas as a function of energy. This analysis considered dose locations external to the 100-ton HI-TRAC transfer cask and the HI-STORM *100* overpack and vents. The results of this analysis have revealed that, due to the magnitude of the gamma source at lower energies, gammas with energies as low as 0.45 MeV must be included in the shielding analysis. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant (less than 1% of the total gamma dose at all high dose locations). This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low (less than 1% of the total source). Therefore, all gammas with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations. Dose rate contributions from above and below this range were evaluated and found to be negligible. Photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Conservatively, the impurity level of ^{59}Co was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the same 1.0 gm/kg impurity level.

Holtec International has gathered information from utilities and vendors which shows that the 1.0 gm/kg impurity level is very conservative for fuel which has been manufactured since the mid-to-late 1980s after the implementation of an industry wide cobalt reduction program. The typical Cobalt-59 impurity level for fuel since the late 1980s is less than 0.5 gm/kg. Based on this, fuel with a short cooling time, 5 to 9 years, would have a Cobalt-59 impurity level less than 0.5 gm/kg. Therefore, the use of a bounding Cobalt-59 impurity level of 1.0 gm/kg is very conservative, particularly for recently manufactured assemblies. Analysis in Reference [5.2.3] indicates that the cobalt impurity in steel and inconel for fuel manufactured in the 1970s ranged from approximately 0.2 gm/kg to 2.2 gm/kg. However, older fuel manufactured with higher cobalt impurity levels will also have a corresponding longer cooling time and therefore will be bounded by the analysis presented in this chapter. As confirmation of this statement, Appendix D presents a comparison of the dose rates around the 100-ton HI-TRAC and the HI-STORM with the MPC-24 for a short cooling time (5 years) using the 1.0 gm/kg mentioned above and for a long cooling time (9 years) using a higher cobalt impurity level of 4.7 gm/kg for inconel. These results confirm that the dose rates for the longer cooling time with the higher impurity level are essentially equivalent to (within 11%) or bounded by the dose rates for the shorter cooling time with the lower impurity level. Therefore, the analysis in this chapter is conservative.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM 100 system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 1 gm/kg (0.1 wt%) was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. The masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative ^{59}Co impurity level and the use of conservative flux weighting fractions (discussed below) results in an over-prediction of the non-fuel hardware source that bounds all fuel for which storage is requested.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the ^{60}Co is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.10. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.1~~2~~*1* ~~and~~ *through* 5.2.13 provide the ^{60}Co activity utilized in the shielding calculations for the non-fuel regions of the assemblies in the *MPC-32,* MPC-24, and the MPC-68 *for varying burnup and cooling times*. The design basis ~~damaged~~ *6x6* and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR ~~intact~~ design basis fuel. This is a conservative assumption as the design basis ~~damaged~~ *6x6* fuel and MOX fuel assemblies are limited to a significantly lower burnup and longer cooling time than the ~~intact~~ design basis fuel.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.24 presents the ^{235}U initial enrichments for various burnup ranges from 20,000 - ~~50~~*70*,000 MWD/MTU for PWR and BWR zircaloy clad fuel. These enrichments are based on Reference*s* [5.2.6] *and [5.2.7]*. Table 8 of ~~this~~ reference *[5.2.6]* presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.24, *for burnups up to 50,000 MWD/MTU,* are approximately the average enrichments *from Table 8 of reference [5.2.6]* for the burnup range that is 5,000 MWD/MTU less than the ranges listed in Table 5.2.24. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. *For burnups greater than 50,000 MWD/MTU, the data on historical and projected burnups available in the LWR Quantities Database in reference [5.2.7] was reviewed and conservatively low enrichments were chosen for each burnup range above 50,000 MWD/MTU.*

Inherent to this approach of selecting minimum enrichments that bound the vast majority of discharged fuel is the fact that a small number of atypical assemblies will not be bounded. However, these atypical assemblies are very few in number (as evidenced by the referenced discharge data), and thus, it is unlikely that a single cask would contain several of these outlying assemblies. Further, because the approach is based on using minimum enrichments for given burnup ranges, any atypical assemblies that may exist are expected to have enrichments that are very near to the minimum enrichments used in the analysis. Therefore, the result is an insignificant effect on the calculated dose rates. Consequently, the minimum enrichment values used in the analysis are adequate to bound the fuel authorized by the ~~technical specifications~~*limits in the CoC* for loading in the HI-STORM system. Therefore a minimum enrichment is not specified in the ~~Technical Specifications~~*limits in the CoC*. Since the enrichment does affect the source term evaluation, it is recommended that the site-specific evaluation under 10CFR72.212 consider the appropriate minimum enrichment for the fuel being stored.

The neutron source calculated for the design basis ~~intact~~ fuel assemblies for the MPC-24, *MPC-32,* and MPC-68 and the design basis ~~damaged~~ *6x6* fuel are listed in Tables 5.2.~~16~~ *15* through 5.2.18 in neutrons/s *for varying burnup and cooling times*. Table 5.2.23 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. ^{244}Cm accounts for approximately 96% of the total number of neutrons produced, with slightly over 2% originating from (α,n) reactions within the UO_2 fuel. The remaining 2% derive from spontaneous fission in various Pu and Cm radionuclides. In addition, any neutrons generated from subcritical multiplication, (n,2n) or similar reactions are properly accounted for in the MCNP calculation.

5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.3 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.3 is actually longer than the true active fuel length of 122 inches for the WE 15x15 and 83 inches for the LaCrosse 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this does not reflect the potential change in dose rates. As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 neutrons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches and 144 inches, respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 neutrons/s/inch and 1 neutron/s/inch for the stainless steel and zircaloy fuel, respectively. The result would be a higher neutron dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

In order to eliminate the potential confusion when comparing source terms, the stainless steel clad fuel source terms were calculated with the same active fuel length as the design basis zircaloy clad fuel. Reference [5.2.2] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level was used for the stainless steel cladding in the source term calculations. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fitting masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.8, 5.2.9, 5.2.19, and 5.2.20 list the gamma and neutron source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV, which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.4 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

5.2.4 Control Components

~~Control components are not permitted for storage in the HI-STORM 100 System.~~*Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM 100 System as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted to the inner four fuel storage locations in the MPC-24, MPC-24E, and the MPC-32.*

5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. Since these devices are *made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.*

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STORM 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.30. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRA.

Once the bounding BPRA and TPD were determined, the allowable Co-60 source from the BPRA and TPD were specified: 50 curies Co-60 for each TPD and 831 curies Co-60 for each BPRA. Table 5.2.31 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). An allowable burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. These burnup and cooling times assure that the Cobalt-60 activity remains below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 *production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.*

Section 5.4.6 discusses the increase in the cask dose rates due to the insertion of BPRAs or TPDs into fuel assemblies.

5.2.4.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B_4C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is being limited to four CRAs and/or APSRs. These four devices are required to be stored in the inner four locations in the MPC-24, MPC-24E, MPC-24EF, and MPC-32 as outlined in Appendix B to the CoC.

In order to determine the impact on the dose rates around the HI-STORM 100 System, source terms for the CRAs and APSRs were calculated using SAS2H and ORIGEN-S. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating 1 kg of steel, inconel, and AgInCd using the flux calculated for the design basis B&W 15x15 fuel assembly. The total curies of cobalt for the steel and inconel and the 0.3-1.0 MeV source for the

AgInCd were calculated as a function of burnup and cooling time to a maximum burnup of 630,000 MWD/MTU. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU. The sources were then scaled by the appropriate mass using the flux weighting factors for the different regions of the assembly to determine the final source term. Two different configurations were analyzed for both the CRAs and APSRs with an additional third configuration analyzed for the APSRs. The configurations, which are summarized below, are described in Tables 5.2.32 for the CRAs and Table 5.2.33 for the APSR. The masses of the materials listed in these tables were determined from a review of [5.2.5] with bounding values chosen. The masses listed in Tables 5.2.32 and 5.2.33 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

Configuration 1: CRA and APSR
This configuration had the lower 15 inches of the CRA and APSR activated at full flux with two regions above the 15 inches activated at a reduced power level. This simulates a CRA or APSR which was operated at 10% insertion. The regions above the 15 inches reflect the upper portion of the fuel assembly.

Configuration 2: CRA and APSR
This configuration represents a fully removed CRA or APSR during normal core operations. The activated portion corresponds to the upper portion of a fuel assembly above the active fuel length with the appropriate flux weighting factors used.

Configuration 3: APSR
This configuration represents a fully inserted gray APSR during normal core operations. The region in full flux was assumed to be the 63 inches of the absorber.

Tables 5.2.34 and 5.2.35 present the source terms that were calculated for the CRAs and APSRs respectively. The only significant source from the activation of inconel or steel is Co-60 and the only significant source from the activation of AgInCd is from 0.3-1.0 MeV. The source terms for CRAs, Table 5.2.34, were calculated for a maximum burnup of 630,000 MWD/MTU and a minimum cooling time of 5 years. Because of the significant source term in APSRs that have seen extensive in-core operations, the source term in Table 5.2.35 was calculated to be a bounding source term for a variable burnup and cooling time as outlined in Appendix B to the CoC. The very larger Cobalt-60 activity in configuration 3 in Table 5.2.35 is due to the assumed Cobalt-59 impurity level of 4.7 gm/kg. If this impurity level were similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

Section 5.4.6 discusses the effect on dose rate of the insertion of APSRs and CRAs into the inner four fuel assemblies in the MPC-24 or MPC-32.

5.2.5 Choice of Design Basis Assembly

The analysis presented in this chapter was performed to bound the fuel assembly classes listed in Tables 2.1.1 and 2.1.2. In order to perform a bounding analysis, a design basis fuel assembly must be chosen. Therefore, a fuel assembly from each fuel class was analyzed and a comparison of the neutrons/sec, photons/sec, and thermal power (watts) was performed. The fuel assembly that produced the highest source for a specified burnup, cooling time, and enrichment was chosen as the design basis fuel assembly. A separate design basis assembly was chosen for the *PWR MPCs* (MPC-24 *and MPC-32)* and the *BWR MPCs* (MPC-68).

5.2.5.1 PWR Design Basis Assembly

Table 2.1.1 lists the PWR fuel assembly classes that were evaluated to determine the design basis PWR fuel assembly. Within each class, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within a class are very minor (pellet diameter, clad thickness, etc.), it is conservative to choose the assembly with the highest UO_2 mass. For a given class of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, the highest UO_2 mass will have produced the most energy and therefore the most fission products.

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. The fuel assembly listed for each class is the assembly with the highest UO_2 mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.25. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the *Indian Point 1,* Haddam Neck, and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time. The initial enrichment used in the analysis is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.27. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.1. This fuel assembly also has the highest UO_2 mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term.

The Haddam Neck and San Onofre 1 classes are shorter stainless steel clad versions of the WE 15x15 and WE 14x14 classes, respectively. Since these assemblies have stainless steel clad, they were analyzed separately as discussed in Section 5.2.3. Based on the results in Table 5.2.27, which show that the WE 15x15 assembly class has a higher source term than the WE 14x14 assembly class, the Haddam Neck, WE 15x15, fuel assembly was analyzed as the bounding PWR stainless steel clad fuel assembly. *The Indian Point 1 fuel assembly is a unique 14x14 design with a smaller mass of fuel and clad than the WE14x14. Therefore, it is also bounded by the WE 15x15 stainless steel fuel assembly.*

5.2.5.2 BWR Design Basis Assembly

Table 2.1.2 lists the BWR fuel assembly classes that were evaluated to determine the design basis BWR fuel assembly. Since there are minor differences between the array types in the GE BWR/2-3 and GE BWR/4-6 assembly classes, these assembly classes were not considered individually but rather as a single class. Within that class, the array types, 7x7, 8x8, 9x9, and 10x10 were analyzed to determine the bounding BWR fuel assembly. Since the Humboldt Bay 7x7 and Dresden 1 8x8 are smaller versions of the 7x7 and 8x8 assemblies they are bounded by the 7x7 and 8x8 assemblies in the GE BWR/2-3 and GE BWR/4-6 classes. Within each array type, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within an array type are very minor, it is conservative to choose the assembly with the highest UO_2 mass. For a given array type of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, it will have produced the most energy and therefore the most fission products. The Humboldt Bay 6x6, Dresden 1 6x6, and LaCrosse assembly classes were not considered in the determination of the bounding fuel assembly. However, these assemblies were analyzed explicitly as discussed below.

Table 5.2.26 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. The fuel assembly listed for each array type is the assembly that has the highest UO_2 mass. All fuel assemblies in Table 5.2.26 were analyzed at the same burnup and cooling time. The initial enrichment used in these analyses is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.28. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.2. This fuel assembly also has the highest UO_2 mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. According to Reference [5.2.6], the last discharge of a 7x7 assembly was in 1985 and the maximum average burnup for a 7x7 during their operation was 29,000 MWD/MTU. This clearly indicates that the existing 7x7 assemblies have an average burnup and minimum cooling time that is well within the burnup and cooling time limits in ~~Technical Specification Table 2.1-4, Appendix 12.A~~*Appendix B to the CoC*. Therefore, the 7x7 assembly has never reached the burnup level analyzed in this chapter. However, in the interest of conservatism the 7x7 was chosen as the bounding fuel assembly array type.

Since the LaCrosse fuel assembly type is a stainless steel clad 10x10 assembly it was analyzed separately. The maximum burnup and minimum cooling time for this assembly are limited to 22,500 MWD/MTU and 10-year cooling as specified in ~~Technical Specification Table 2.1-1, Appendix 12.A~~*Appendix B to the CoC*. This assembly type is discussed further in Section 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter fuel than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes because it has the higher UO_2 mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies, which were explicitly analyzed as well.

Reference [5.2.6] indicates that the Dresden 1 6x6 fuel assembly has a higher UO_2 mass than the Dresden 1 8x8 or the Humboldt Bay fuel (6x6 and 7x7). Therefore, the Dresden 1 6x6 fuel assembly was also chosen as the bounding assembly for damaged fuel and fuel debris for the Humboldt Bay and Dresden 1 fuel assembly classes.

Since the design basis ~~damaged fuel assembly and the design basis intact~~ 6x6 fuel assembly *can be intact or damaged*~~are identical~~, the analysis presented in Section 5.4.2 for the damaged *6x6* fuel assembly also demonstrates the acceptability of storing intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

5.2.5.3 Decay Heat Loads

Section 2.1.6 describes the calculation of the burnup versus cooling time ~~Technical Specification~~*limits in the CoC* that ~~is~~ *are* based on a maximum permissible decay heat per assembly. The decay heat values per assembly were calculated using the methodology described in Section 5.2. The design basis fuel assemblies, as described in Table 5.2.1, were used in the calculation of the burnup versus cooling time ~~Technical Specification~~*limits in the CoC*. The enrichments used in the calculation of the decay heats were consistent with Table 5.2.24. As demonstrated in Tables 5.2.27 and 5.2.28, the design basis fuel assembly produces a higher decay heat value than the other assembly types considered. This is due to the higher heavy metal mass in the design basis fuel assemblies. Conservatively, ~~the Technical Specifications~~*Appendix B to the CoC* limits the heavy metal mass to a value less than the design basis value utilized in this chapter. This provides additional assurance that the decay heat values are bounding values.

As further demonstration that the decay heat values (calculated using the design basis fuel assemblies) are conservative, a comparison between these calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.29. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

The heavy metal mass of the non-design basis fuel assembly classes in Appendix B of the Certificate of Compliance are limited to the masses used in Tables 5.2.25 and 5.2.26. No margin is applied between the allowable mass and the analyzed mass of heavy metal for the non-design basis fuel assemblies. This is acceptable because additional assurance that the decay heat values for the non-design basis fuel assemblies are bounding values is obtained by using the decay heat values for the design basis fuel assemblies to determine the acceptable storage criteria for all fuel assemblies. As mentioned above, Table 5.2.29 demonstrates the level of conservatism in applying the decay heat from the design basis fuel assembly to all fuel assemblies.

As mentioned above, the fuel assembly burnup and cooling times in Appendix B to the CoC were calculated using the decay heat limits which are also stipulated in Appendix B to the CoC. The *burnup and cooling times for the non-fuel hardware, in Appendix B to the CoC, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup and cooling times were calculated without consideration for the decay heat from BPRAs, TPDs, CRAs, or APSRs. This is acceptable since the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in Appendix B to the CoC regardless of the heat source (assembly or non-fuel hardware) and the actual decay heat from the non-fuel hardware is expected to be minimal. In addition, the shielding analysis presented in this chapter conservatively calculates the dose rates using both the burnup and cooling times for the fuel assemblies and non-fuel hardware. Therefore, the safety of the HI-STORM 100 system is guaranteed through the bounding analysis in this chapter, represented by the burnup and cooling time limits in the CoC, and the bounding thermal analysis in Chapter 4, represented by the decay heat limits in the CoC.*

5.2.6 Thoria Rod Canister

Dresden Unit 1 has a single DFC containing 18 thoria rods which have obtained a relatively low burnup, 16,000 MWD/MTU. These rods were removed from two 8x8 fuel assemblies which contained 9 rods each. The irradiation of thorium produces an isotope which is not commonly found in depleted uranium fuel. Th-232 when irradiated produces U-233. The U-233 can undergo an (n,2n) reaction which produces U-232. The U-232 decays to produce Tl-208 which produces a 2.6 MeV gamma during Beta decay. This results in a significant source in the 2.5-3.0 MeV range which is not commonly present in depleted uranium fuel. Therefore, this single DFC container was analyzed to determine if it was bounded by the current shielding analysis.

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.36 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Section 5.4.8 provides a further discussion of the thoria rod canister and its acceptability for storage in the HI-STORM 100 System.

5.2.7 Fuel Assembly Neutron Sources

Neutron sources are used in reactors during initial startup of reactor cores. There a different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Section 5.4.7 to demonstrate that they are acceptable for storage in the HI-STORM 100 System. Currently these are the only neutron source permitted for storage in the HI-STORM 100 System.

5.2.8 Stainless Steel Channels

The LaCrosse nuclear plant used two types of channels for their BWR assemblies: stainless steel and zircaloy. Since the irradiation of zircaloy does not produce significant activation, there are no restrictions on the storage of these channels and they are not explicitly analyzed in this chapter. The stainless steel channels, however, can produce a significant amount of activation, predominantly from Co-60. LaCrosse has thirty-two stainless steel channels, a few of which, have been in the reactor core for, approximately, the lifetime of the plant. Therefore, the activation of the stainless steel channels was conservatively calculated to demonstrate that they are acceptable for storage in the HI-STORM 100 system. For conservatism, the number of stainless steel channels in an MPC-68 is being limited to sixteen and Appendix B to the CoC requires that these channels be stored in the inner sixteen locations.

The activation of a single stainless steel channel was calculated by simulating the irradiation of the channels with ORIGEN-S using the flux calculated from the LaCrosse fuel assembly. The mass of the steel channel in the active fuel zone (83 inches) was used in the analysis. For burnups beyond 22,500 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 22,500 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 22,500 MWD/MTU.

LaCrosse was commercially operated from November 1969 until it was shutdown in April 1987. Therefore, the shortest cooling time for the assemblies and the channels is 13 years. Assuming the plant operated continually from 11/69 until 4/87, approximately 17.5 years or 6388 days, the accumulated burnup for the channels would be 186,000 MWD/MTU (6388 days times 29.17 MW/MTU from Table 5.2.3). Therefore, the cobalt activity calculated for a single stainless steel channel irradiated for 180,000 MWD/MTU was calculated to be 667 curies of Co-60 for 13 years cooling. This is equivalent to a source of 4.94E+13 photons/sec in the energy range of 1.0-1.5 MeV.

In order to demonstrate that sixteen stainless steel channels are acceptable for storage in an MPC-68, a comparison of source terms is performed. Table 5.2.8 indicates that the source term for the LaCrosse design basis fuel assembly in the 1.0-1.5 MeV range is 6.34E+13 photons/sec for 10 years cooling, assuming a 144 inch active fuel length. This is equivalent to 4.31E+15 photons/sec/cask. At 13 years cooling, the fuel source term in that energy range decreases to 4.31E+13 photons/sec which is equivalent to 2.93E+15 photons/sec/cask. If the source term from the stainless steel channels is scaled to 144 inches and added to the 13 year fuel source term the result is 4.30E+15 photons/sec/cask (2.93E+15 photons/sec/cask + 4.94E+13 photons/sec/channel x 144 inch/83 inch x 16 channels/cask). This number is equivalent to the 10 *year 4.31E+15 photons/sec/cask source calculated from Table 5.2.8 and used in the shielding analysis in this chapter. Therefore, it is concluded that the storage of 16 stainless steel channels in an MPC-68 is acceptable.*

Table 5.2.1

DESCRIPTION OF DESIGN BASIS ~~INTACT~~ ZIRCALOY CLAD FUEL

	PWR	BWR
Assembly type/class	B&W 15x15	GE 7x7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
Burnup (MWD/MTU)†	*52,500 (MPC-24)* 45,000 (~~MPC-24~~*MPC-32*)	~~45,000~~*47,500* (MPC-68)
Cooling Time (years)†	5 (MPC-24 *and 32*)	5 (MPC-68)
Specific power (MW/MTU)	40	30
Weight of UO_2 (kg)††	562.029	225.177
Weight of U (kg)††	495.485	198.516

Notes:

1. The B&W 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun.
2. The GE 7x7 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8.

† Burnup and cooling time combinations conservatively bound the acceptable burnup and cooling times listed in ~~Technical Specification Table 2.1-4, Appendix 12.A~~*Appendix B to the CoC.*

†† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS ~~INTACT~~ FUEL

	PWR	BWR
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2

DESCRIPTION OF DESIGN BASIS ~~DAMAGED~~ *GE 6x6* ZIRCALOY CLAD FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.694
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.035
Pellet diameter (in.)	0.494
Pellet material	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2 (kg)†	129.5
Weight of U (kg)†	114.2

Notes:
1. The 6x6 is the design basis damaged fuel assembly for the Humboldt Bay (all array types) and the Dresden 1 (all array types) damaged fuel assembly classes. It is also the design basis fuel assembly for the intact Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes.
2. This design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.

† Derived from parameters in this table.

Table 5.2.3

DESCRIPTION OF DESIGN BASIS ~~INTACT~~ STAINLESS STEEL CLAD FUEL

	PWR	BWR
Fuel type	WE 15x15	LaCrosse 10x10
Active fuel length (in.)	144	144
No. of fuel rods	204	100
Rod pitch (in.)	0.563	0.565
Cladding material	304 SS	348H SS
Rod diameter (in.)	0.422	0.396
Cladding thickness (in.)	0.0165	0.02
Pellet diameter (in.)	0.3825	0.35
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.5	3.5
Burnup (MWD/MTU)†	40,000 (MPC-24 *and 32*)	22,500 (MPC-68)
Cooling Time (years)†	8 (MPC-24), *9 (MPC-32)*	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.)	0.546	N/A
Water Rod Thickness (in.)	0.017	N/A

Notes:

1. The WE 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: *Indian Point 1,* Haddam Neck, and San Onofre 1.
2. The LaCrosse 10x10 is the design basis assembly for the following fuel assembly class listed in Table 2.1.2: LaCrosse.

† *Burnup and cooling time combinations are equivalent to or conservatively bound the limits in Appendix B to the CoC.*

Table 5.2.4

CALCULATED MPC-32 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	***Upper Energy***	***32,500 MWD/MTU 5 Year Cooling***		***45,000 MWD/MTU 5 Year Cooling***		***45,000 MWD/MTU 10 Year Cooling***	
(MeV)	*(MeV)*	*(MeV/s)*	*(Photons/s)*	*(MeV/s)*	*(Photons/s)*	*(MeV/s)*	*(Photons/s)*
0.45	*0.7*	*1.47E+15*	*2.56E+15*	*2.09E+15*	*3.63E+15*	*1.33E+15*	*2.32E+15*
0.7	*1.0*	*4.49E+14*	*5.28E+14*	*7.06E+14*	*8.31E+14*	*1.62E+14*	*1.91E+14*
1.0	*1.5*	*1.07E+14*	*8.53E+13*	*1.62E+14*	*1.30E+14*	*6.79E+13*	*5.43E+13*
1.5	*2.0*	*7.51E+12*	*4.29E+12*	*9.97E+12*	*5.70E+12*	*3.35E+12*	*1.92E+12*
2.0	*2.5*	*6.42E+12*	*2.86E+12*	*7.06E+12*	*3.14E+12*	*1.34E+11*	*5.97E+10*
2.5	*3.0*	*2.38E+11*	*8.67E+10*	*2.89E+11*	*1.05E+11*	*1.02E+10*	*3.71E+09*
Total		*2.04E+15*	*3.18E+15*	*2.97E+15*	*4.60E+15*	*1.57E+15*	*2.57E+15*

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Table 5.2.5

CALCULATED MPC-24 PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	***42,500* MWD/MTU 5 Year Cooling**		***52,500* MWD/MTU 5 Year Cooling**		***57,500* MWD/MTU *12* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*1.97E+15*	*3.42E+15*	*2.47E+15*	*4.29E+15*	*1.55E+15*	*2.69E+15*
0.7	1.0	*6.54E+14*	*7.70E+14*	*8.78E+14*	*1.03E+15*	*1.36E+14*	*1.61E+14*
1.0	1.5	*1.51E+14*	*1.21E+14*	*1.99E+14*	*1.59E+14*	*7.44E+13*	*5.95E+13*
1.5	2.0	*9.51E+12*	*5.43E+12*	*1.15E+13*	*6.56E+12*	*3.82E+12*	*2.18E+12*
2.0	2.5	*6.97E+12*	*3.10E+12*	*7.29E+12*	*3.24E+12*	*4.16E+10*	*1.85E+10*
2.5	3.0	*2.82E+11*	*1.03E+11*	*3.17E+11*	*1.15E+11*	*4.17E+09*	*1.52E+09*
Total		*2.79E+15*	*4.32E+15*	*3.56E+15*	*5.49E+15*	*1.76E+15*	*2.91E+15*

Table 5.2.6

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	***40,000* MWD/MTU 5 Year Cooling**		***47,500* MWD/MTU 5 Year Cooling**		***50,000* MWD/MTU *10* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*7.15E+14*	*1.24E+15*	*8.58E+14*	*1.49E+15*	*5.82E+14*	*1.01E+15*
0.7	1.0	*2.25E+14*	*2.64E+14*	*2.85E+14*	*3.36E+14*	*6.82E+13*	*8.03E+13*
1.0	1.5	*5.14E+13*	*4.11E+13*	*6.38E+13*	*5.10E+13*	*2.82E+13*	*2.25E+13*
1.5	2.0	*3.18E+12*	*1.82E+12*	*3.69E+12*	*2.11E+12*	*1.38E+12*	*7.90E+11*
2.0	2.5	*2.19E+12*	*9.75E+11*	*2.26E+12*	*1.00E+12*	*4.57E+10*	*2.03E+10*
2.5	3.0	*9.40E+10*	*3.42E+10*	*1.05E+11*	*3.82E+10*	*3.72E+09*	*1.35E+09*
Total		*9.96E+14*	*1.55E+15*	*1.21E+15*	*1.88E+15*	*6.79E+14*	*1.12E+15*

Table 5.2.7

CALCULATED MPC-68 ~~AND MPC-68F~~ BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD ~~DAMAGED~~ *GE 6x6* FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.53e+14	2.65e+14
7.0e-01	1.0	3.97e+12	4.67e+12
1.0	1.5	3.67e+12	2.94e+12
1.5	2.0	2.20e+11	1.26e+11
2.0	2.5	1.35e+09	5.99e+08
2.5	3.0	7.30e+07	2.66e+07
Totals		1.61e+14	2.73e+14

Table 5.2.8

CALCULATED BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	Upper Energy	22,500 MWD/MTU 10-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	2.72e+14	4.74+14
7.0e-01	1.0	1.97e+13	2.31e+13
1.0	1.5	7.93e+13	6.34e+13
1.5	2.0	4.52e+11	2.58e+11
2.0	2.5	3.28e+10	1.46e+10
2.5	3.0	1.69e+9	6.14e+8
Totals		3.72e+14	5.61e+14

Note: These source terms were calculated for a 144-inch fuel length. The ~~Technical Specification~~*limits* in ~~Chapter 12~~*Appendix B to the CoC* ~~is~~ *are* based on the actual 83-inch active fuel length.

Table 5.2.9

CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	**Upper Energy**	**40,000 MWD/MTU 8-Year Cooling**		***40,000 MWD/MTU 9-Year Cooling***	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	*(MeV/s)*	*(Photons/s)*
4.5e-01	7.0e-01	1.37e+15	2.38e+15	*1.28E+15*	*2.22E+15*
7.0e-01	1.0	2.47e+14	2.91e+14	*1.86E+14*	*2.19E+14*
1.0	1.5	4.59e+14	3.67e+14	*4.02E+14*	*3.21E+14*
1.5	2.0	3.99e+12	2.28e+12	*3.46E+12*	*1.98E+12*
2.0	2.5	5.85e+11	2.60e+11	*2.69E+11*	*1.20E+11*
2.5	3.0	3.44e+10	1.25e+10	*1.77E+10*	*6.44E+09*
Totals		2.08e+15	3.04e+15	*1.87E+15*	*2.76E+15*

Note: These source terms were calculated for a 144-inch fuel length. The ~~Technical Specification~~*limits* in ~~Chapter 12~~*Appendix B to the CoC* ~~is~~ *are* based on the actual 122-inch active fuel length.

Table 5.2.10

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

Region	PWR	BWR
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 5.2.11

CALCULATED MPC-32 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL AT DESIGN BASIS BURNUP AND COOLING TIME

Location	***32,500 MWD/MTU and 5-Year Cooling (curies)***	***45,000 MWD/MTU and 5-Year Cooling (curies)***	***45,000 MWD/MTU and 10-Year Cooling (curies)***
Lower End Fitting	*139.25*	*167.06*	*86.46*
Gas Plenum Springs	*10.62*	*12.75*	*6.60*
Gas Plenum Spacer	*6.10*	*7.31*	*3.79*
Expansion Springs	*N/A*	*N/A*	*N/A*
Incore Grid Spacers	*360.64*	*432.67*	*223.93*
Upper End Fitting	*68.30*	*81.94*	*42.41*
Handle	*N/A*	*N/A*	*N/A*

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Table 5.2.12

CALCULATED MPC-24 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL AT DESIGN BASIS BURNUP AND COOLING TIME

Location	***42,500* MWD/MTU and 5-Year Cooling (curies)**	***52,500* MWD/MTU and 5-Year Cooling (curies)**	***57,500* MWD/MTU and - *12* Year Cooling (curies)**
Lower End Fitting	*163.47*	*183.33*	*76.06*
Gas Plenum Springs	*12.47*	*13.99*	*5.80*
Gas Plenum Spacer	*7.16*	*8.03*	*3.33*
Expansion Springs	N/A	N/A	N/A
Incore Grid Spacers	*423.36*	*474.81*	*196.98*
Upper End Fitting	*80.18*	*89.92*	*37.31*
Handle	N/A	N/A	N/A

Table 5.2.13

CALCULATED MPC-68 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS
ZIRCALOY CLAD FUEL
AT DESIGN BASIS BURNUP AND COOLING TIME

Location	***40,000* MWD/MTU and 5-Year Cooling (curies)**	***47,500* MWD/MTU and 5-Year Cooling (curies)**	***50,000* MWD/MTU and - *10* Year Cooling (curies)**
Lower End Fitting	*63.49*	*71.35*	*36.00*
Gas Plenum Springs	*19.40*	*21.80*	*11.00*
Gas Plenum Spacer	N/A	N/A	N/A
Expansion Springs	*3.53*	*3.96*	*2.00*
Grid Spacer Springs	*29.10*	*32.70*	*16.50*
Upper End Fitting	*17.64*	*19.82*	*10.00*
Handle	*2.20*	*2.48*	*1.25*

Table 5.2.14

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Table 5.2.15

CALCULATED MPC-32 PWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	***Upper Energy (MeV)***	***32,500 MWD/MTU 5-Year Cooling (Neutrons/s)***	***45,000 MWD/MTU 5-Year Cooling (Neutrons/s)***	***45,000 MWD/MTU 10-Year Cooling (Neutrons/s)***
1.0e-01	*4.0e-01*	*6.35E+06*	*1.63E+07*	*1.35E+07*
4.0e-01	*9.0e-01*	*3.24E+07*	*8.33E+07*	*6.89E+07*
9.0e-01	*1.4*	*2.98E+07*	*7.63E+07*	*6.31E+07*
1.4	*1.85*	*2.20E+07*	*5.62E+07*	*4.66E+07*
1.85	*3.0*	*3.90E+07*	*9.92E+07*	*8.25E+07*
3.0	*6.43*	*3.52E+07*	*9.01E+07*	*7.46E+07*
6.43	*20.0*	*3.11E+06*	*7.98E+06*	*6.60E+06*
Totals		*1.68E+08*	*4.29E+08*	*3.56E+08*

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Table 5.2.16

CALCULATED MPC-24 PWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	***42,500* MWD/MTU 5-Year Cooling (Neutrons/s)**	***52,500* MWD/MTU 5-Year Cooling (Neutrons/s)**	***57,500* MWD/MTU *12*-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	*1.42E+07*	*2.64E+07*	*2.52E+07*
4.0e-01	9.0e-01	*7.26E+07*	*1.35E+08*	*1.29E+08*
9.0e-01	1.4	*6.65E+07*	*1.24E+08*	*1.18E+08*
1.4	1.85	*4.90E+07*	*9.09E+07*	*8.69E+07*
1.85	3.0	*8.66E+07*	*1.60E+08*	*1.54E+08*
3.0	6.43	*7.86E+07*	*1.46E+08*	*1.39E+08*
6.43	20.0	*6.96E+06*	*1.29E+07*	*1.24E+07*
Totals		*3.75E+08*	*6.95E+08*	*6.64E+08*

Table 5.2.17

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	***40,000* MWD/MTU 5-Year Cooling (Neutrons/s)**	***47,500* MWD/MTU 5-Year Cooling (Neutrons/s)**	***50,000* MWD/MTU *10*-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	*5.03E+06*	*9.02E+06*	*7.43E+06*
4.0e-01	9.0e-01	*2.57E+07*	*4.61E+07*	*3.80E+07*
9.0e-01	1.4	*2.35E+07*	*4.22E+07*	*3.48E+07*
1.4	1.85	*1.73E+07*	*3.11E+07*	*2.56E+07*
1.85	3.0	*3.06E+07*	*5.47E+07*	*4.52E+07*
3.0	6.43	*2.78E+07*	*4.98E+07*	*4.11E+07*
6.43	20.0	*2.46E+06*	*4.42E+06*	*3.64E+06*
Totals		*1.32E+08*	*2.37E+08*	*1.96E+08*

Table 5.2.18

CALCULATED MPC-68 ~~AND MPC-68F~~ BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD ~~DAMAGED~~ *GE 6x6* FUEL

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	8.22e+5
4.0e-01	9.0e-01	4.20e+6
9.0e-01	1.4	3.87e+6
1.4	1.85	2.88e+6
1.85	3.0	5.18e+6
3.0	6.43	4.61e+6
6.43	20.0	4.02e+5
Total		2.20e+7

Table 5.2.19

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**22,500 MWD/MTU 10-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.23e+5
4.0e-01	9.0e-01	1.14e+6
9.0e-01	1.4	1.07e+6
1.4	1.85	8.20e+5
1.85	3.0	1.56e+6
3.0	6.43	1.30e+6
6.43	20.0	1.08e+5
Total		6.22e+6

Note: These source terms were calculated for a 144-inch fuel length. The ~~Technical Specification~~*limits* in ~~Chapter 12~~*Appendix B to the CoC* ~~is~~ *are* based on the actual 83-inch active fuel length.

Table 5.2.20

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**40,000 MWD/MTU 8-Year Cooling (Neutrons/s)**	***40,000 MWD/MTU 9-Year Cooling (Neutrons/s)***
1.0e-01	4.0e-01	1.04e+7	*1.01E+07*
4.0e-01	9.0e-01	5.33e+7	*5.14E+07*
9.0e-01	1.4	4.89e+7	*4.71E+07*
1.4	1.85	3.61e+7	*3.48E+07*
1.85	3.0	6.41e+7	*6.18E+07*
3.0	6.43	5.79e+7	*5.58E+07*
6.43	20.0	5.11e+6	*4.92E+06*
Totals		2.76e+8	*2.66E+08*

Note: These source terms were calculated for a 144-inch fuel length. The ~~Technical Specification~~*limits* in ~~Chapter 12~~*Appendix B to the CoC* ~~is~~ *are* based on the actual 122-inch active fuel length.

Table 5.2.21

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.696
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.036
Pellet diameter (in.)	0.482
Pellet material	UO_2 and $PuUO_2$
No. of UO_2 Rods	27
No. of $PuUO_2$ rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)†	2.24 (UO_2 rods) 0.711 ($PuUO_2$ rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2,$PuUO_2$ (kg)††	123.3
Weight of U,Pu (kg)††	108.7

† See Table 5.3.3 for detailed composition of $PuUO_2$ rods.

†† Derived from parameters in this table.

Table 5.2.22

CALCULATED MPC-68 ~~AND MPC-68F~~ BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.45e+14	2.52e+14
7.0e-01	1.0	3.87e+12	4.56e+12
1.0	1.5	3.72e+12	2.98e+12
1.5	2.0	2.18e+11	1.25e+11
2.0	2.5	1.17e+9	5.22e+8
2.5	3.0	9.25e+7	3.36e+7
Totals		1.53e+14	2.60e+14

Table 5.2.23

CALCULATED MPC-68 ~~AND MPC-68F~~ BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**30,000 MWD/MTU 18-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	1.24e+6
4.0e-01	9.0e-01	6.36e+6
9.0e-01	1.4	5.88e+6
1.4	1.85	4.43e+6
1.85	3.0	8.12e+6
3.0	6.43	7.06e+6
6.43	20.0	6.07e+5
Totals		3.37e+7

Table 5.2.24

INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

Burnup Range (MWD/MTU)	Initial Enrichment (wt.% ^{235}U)
BWR Fuel	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
50,000-55,000	*3.6*
55,000-60,000	*4.0*
60,000-65,000	*4.4*
PWR Fuel	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6
50,000-55,000	*3.9*
55,000-60,000	*4.2*
60,000-65,000	*4.5*
65,000-70,000	*4.8*

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc.

Table 5.2.25

DESCRIPTION OF EVALUATED ~~INTACT~~ ZIRCALOY CLAD PWR FUEL

Assembly class	WE 14×14	WE 15×15	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Active fuel length (in.)	144	144	144	144	150	144	144
No. of fuel rods	179	204	264	176	236	208	264
Rod pitch (in.)	0.556	0.563	0.496	0.580	0.5063	0.568	0.502
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.422	0.422	0.374	0.440	0.382	0.428	0.377
Cladding thickness (in.)	0.0243	0.0245	0.0225	0.0280	0.0250	0.0230	0.0220
Pellet diameter (in.)	0.3659	0.366	0.3225	0.377	0.3255	0.3742	0.3252
Pellet material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (95% of theoretical)	10.412	10.412	10.412	10.412	10.412	10.412	10.412
Enrichment (wt.% ^{235}U)	3.4	3.4	3.4	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5	5	5
Specific power (MW/MTU)	40	40	40	40	40	40	40
Weight of UO_2 (kg)[†]	462.451	527.327	529.848	482.706	502.609	562.029	538.757
Weight of U (kg)[†]	407.697	464.891	467.114	425.554	443.100	495.485	474.968
No. of Guide Tubes	17	21	25	5	5	17	25
Guide Tube O.D. (in.)	0.539	0.546	0.474	1.115	0.98	0.53	0.564
Guide Tube Thickness (in.)	0.0170	0.0170	0.0160	0.0400	0.0400	0.0160	0.0175

† Derived from parameters in this table.

Table 5.2.26

DESCRIPTION OF EVALUATED ~~INTACT~~ ZIRCALOY CLAD BWR FUEL

Array Type	7×7	8×8	9×9	10×10
Active fuel length (in.)	144	144	144	144
No. of fuel rods	49	~~63~~*64*	74	92
Rod pitch (in.)	0.738	0.~~640~~*642*	0.566	0.510
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.570	0.~~493~~*484*	0.440	0.404
Cladding thickness (in.)	0.0355	0.~~0340~~*02725*	0.0280	0.0260
Pellet diameter (in.)	0.488	0.~~416~~*4195*	0.376	0.345
Pellet material	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (~~95%~~ of theoretical)	10.412 *(95%)*	10.412 *(95%)*	10.~~412~~*5216* *(96%)*	10.~~412~~*5216* *(96%)*
Enrichment (wt.% ^{235}U)	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
Specific power (MW/MTU)	30	30	30	30
Weight of UO_2 (kg)†	225.177	~~210.385~~*217.336*	~~201.881~~*204.006*	~~211.307~~*213.531*
Weight of U (kg)†	198.516	~~185.475~~*191.603*	~~177.978~~*179.852*	~~186.288~~*188.249*
No. of Water Rods	0	~~1~~*0*	2	2
Water Rod O.D. (in.)	n/a	~~0.493~~*n/a*	0.980	0.980
Water Rod Thickness (in.)	n/a	~~0.0340~~*n/a*	0.0300	0.0300

† Derived from parameters in this table.

Table 5.2.27

COMPARISON OF SOURCE TERMS FOR ~~INTACT~~ ZIRCALOY CLAD PWR FUEL
3.4 wt.% ^{235}U - 40,000 MWD/MTU - 5 years cooling

Assembly class	WE 14×14	WE 15×15	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Neutrons/sec	2.29e+8 / 2.28e+8	2.63e+8 / 2.65e+8	2.62e+8	2.31e+8	2.34e+8	2.94e+8	2.64e+8
Photons/sec (0.45-3.0 MeV)	3.28e+15/ 3.32e+15	3.74e+15/ 3.79e+15	3.76e+15	3.39e+15	3.54e+15	4.01e+15	3.82e+15
Thermal power (watts)	926.6 / 934.9	1056 / 1068	1062	956.6	995.7	1137	1077

Note:
The WE 14x14 and WE 15x15 have both zircaloy and stainless steel guide tubes. The first value presented is for the assembly with zircaloy guide tubes and the second value is for the assembly with stainless steel guide tubes.

Table 5.2.28

COMPARISON OF SOURCE TERMS FOR ~~INTACT~~ ZIRCALOY CLAD BWR FUEL
3.0 wt.% ^{235}U - 40,000 MWD/MTU - 5 years cooling

Assembly class	7×7	8×8	9×9	10×10
Neutrons/sec	1.33e+8	1.~~17e~~*22e*+8	1.~~11e~~*13e*+8	1.~~22e~~*24e*+8
Photons/sec (0.45-3.0 MeV)	1.55e+15	1.~~44e~~*49e*+15	1.~~38e~~*40e*+15	1.~~46e~~*47e*+15
Thermal power (watts)	435.5	~~402.3~~*417.3*	~~385.3~~*389.4*	~~407.4~~*411.5*

Table 5.2.29

COMPARISON OF CALCULATED DECAY HEATS FOR DESIGN BASIS FUEL AND VALUES REPORTED IN THE DOE CHARACTERISTICS DATABASE† FOR 30,000 MWD/MTU AND 5-YEAR COOLING

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from Design Basis Fuel (watts/assembly)
PWR Fuel		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	827.5
CE 16x16	653.7	827.5
CE 14x14	601.3	827.5
WE 17x17	742.5	827.5
WE 15x15	762.2	827.5
WE 14x14	649.6	827.5
BWR Fuel		
7x7	310.9	315.7
8x8	296.6	315.7
9x9	275.0	315.7

Notes:

1. The PWR and BWR design basis fuels are the B&W 15x15 and the GE 7x7, respectively.
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.28, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.

† Reference [5.2.7].

Table 5.2.30

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE

Region	***BPRA***	***TPD***
Upper End Fitting (kg of steel)	*2.62*	*2.3*
Upper End Fitting (kg of inconel)	*0.42*	*0.42*
Gas Plenum Spacer (kg of steel)	*0.77488*	*1.71008*
Gas Plenum Springs (kg of steel)	*0.67512*	*1.48992*
In-core (kg of steel)	*13.2*	*N/A*

Table 5.2.31

DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES

Region	***BPRA***	***TPD***
Upper End Fitting (curies Co-60)	*30.4*	*25.21*
Gas Plenum Spacer (curies Co-60)	*4.6*	*9.04*
Gas Plenum Springs (curies Co-60)	*8.2*	*15.75*
In-core (curies Co-60)	*787.8*	*N/A*

Table 5.2.32

DESCRIPTION OF DESIGN BASIS CONTROL ROD ASSEMBLY CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			***Flux Weighting Factor***	***Mass of cladding (kg Inconel)***	***Mass of absorber (kg AgInCd)***
Start (in)	***Finish (in)***	***Length (in)***			
Configuration 1 - 10% Inserted					
0.0	*15.0*	*15.0*	*1.0*	*1.32*	*7.27*
15.0	*18.8125*	*3.8125*	*0.2*	*0.34*	*1.85*
18.8125	*28.25*	*9.4375*	*0.1*	*0.83*	*4.57*
Configuration 2 - Fully Removed					
0.0	*3.8125*	*3.8125*	*0.2*	*0.34*	*1.85*
3.8125	*13.25*	*9.4375*	*0.1*	*0.83*	*4.57*

Table 5.2.33

DESCRIPTION OF DESIGN BASIS AXIAL POWER SHAPING ROD CONFIGURATION S FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			***Flux Weighting Factor***	***Mass of cladding (kg Steel)***	***Mass of absorber (kg Inconel)***
Start (in)	***Finish (in)***	***Length (in)***			
Configuration 1 - 10% Inserted					
0.0	*15.0*	*15.0*	*1.0*	*1.26*	*5.93*
15.0	*18.8125*	*3.8125*	*0.2*	*0.32*	*1.51*
18.8125	*28.25*	*9.4375*	*0.1*	*0.79*	*3.73*
Configuration 2 - Fully Removed					
0.0	*3.8125*	*3.8125*	*0.2*	*0.32*	*1.51*
3.8125	*13.25*	*9.4375*	*0.1*	*0.79*	*3.73*
Configuration 3 - Fully Inserted					
0.0	*63.0*	*63.0*	*1.0*	*5.29*	*24.89*
63.0	*66.8125*	*3.8125*	*0.2*	*0.32*	*1.51*
66.8125	*76.25*	*9.4375*	*0.1*	*0.79*	*3.73*

Table 5.2.34

DESIGN BASIS SOURCE TERMS FOR CONTROL ROD ASSEMBLY CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			***Photons/sec from AgInCd***			***Curies Co-60 from Inconel***
Start (in)	***Finish (in)***	***Length (in)***	***0.3-0.45 MeV***	***0.45-0.7 MeV***	***0.7-1.0 MeV***	
Configuration 1 - 10% Inserted						
0.0	*15.0*	*15.0*	*1.91e+14*	*1.78e+14*	*1.42e+14*	*1111.38*
15.0	*18.8125*	*3.8125*	*9.71e+12*	*9.05e+12*	*7.20e+12*	*56.50*
18.8125	*28.25*	*9.4375*	*1.20e+13*	*1.12e+13*	*8.92e+12*	*69.92*
Configuration 2 - Fully Removed						
0.0	*3.8125*	*3.8125*	*9.71e+12*	*9.05e+12*	*7.20e+12*	*56.50*
3.8125	*13.25*	*9.4375*	*1.20e+13*	*1.12e+13*	*8.92e+12*	*69.92*

Table 5.2.35

DESIGN BASIS SOURCE TERMS FROM AXIAL POWER SHAPING ROD CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			
Start (in)	***Finish (in)***	***Length (in)***	***Curies of Co-60***
Configuration 1 - 10% Inserted			
0.0	*15.0*	*15.0*	*2682.57*
15.0	*18.8125*	*3.8125*	*136.36*
18.8125	*28.25*	*9.4375*	*168.78*
Configuration 2 - Fully Removed			
0.0	*3.8125*	*3.8125*	*136.36*
3.8125	*13.25*	*9.4375*	*168.78*
Configuration 3 - Fully Inserted			
0.0	*63.0*	*63.0*	*11266.80*
63.0	*66.8125*	*3.8125*	*136.36*
66.8125	*76.25*	*9.4375*	*168.78*

Table 5.2.36

DESCRIPTION OF FUEL ASSEMBLY USED TO ANNALYZE THORIA RODS IN THE THORIA ROD CANISTER

	BWR
Fuel type	8x8
Active fuel length (in.)	110.5
No. of UO_2 fuel rods	55
No. of UO_2/ThO_2 fuel rods	9
Rod pitch (in.)	0.523
Cladding material	zircaloy
Rod diameter (in.)	0.412
Cladding thickness (in.)	0.025
Pellet diameter (in.)	0.358
Pellet material	98.2% ThO_2 and 1.8% UO_2 for UO_2/ThO_2 rods
Pellet density (gm/cc)	10.412
Enrichment (w/o ^{235}U)	93.5 in UO_2 for UO_2/ThO_2 rods and 1.8 for UO_2 rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of THO_2 and UO_2 (kg)†	121.46
Weight of U (kg)†	92.29
Weight of Th (kg)†	14.74

† Derived from parameters in this table.

Table 5.2.37

CALCULATED FUEL GAMMA SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy	***Upper Energy***	***16,000 MWD/MTIHM 18-Year Cooling***	
(MeV)	*(MeV)*	*(MeV/s)*	*(Photons/s)*
4.5e-01	*7.0e-01*	*3.07e+13*	*5.34e+13*
7.0e-01	*1.0*	*5.79e+11*	*6.81e+11*
1.0	*1.5*	*3.79e+11*	*3.03e+11*
1.5	*2.0*	*4.25e+10*	*2.43e+10*
2.0	*2.5*	*4.16e+8*	*1.85e+8*
2.5	*3.0*	*2.31e+11*	*8.39e+10*
Totals		*1.23e+12*	*1.09e+12*

Table 5.2.38

CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy (MeV)	***Upper Energy (MeV)***	***16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)***
1.0e-01	*4.0e-01*	*5.65e+2*
4.0e-01	*9.0e-01*	*3.19e+3*
9.0e-01	*1.4*	*6.79e+3*
1.4	*1.85*	*1.05e+4*
1.85	*3.0*	*3.68e+4*
3.0	*6.43*	*1.41e+4*
6.43	*20.0*	*1.60e+2*
Totals		*7.21e+4*

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM 100 System was performed with MCNP-4A [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STORM 100 System, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Section 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Section 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the Design Drawings that describe the HI-STORM 100 System, including the HI-TRAC transfer casks. These drawings, *using nominal dimensions,* were used to create the MCNP models used in the radiation transport calculations. Figures *5.3.1 through 5.3.6* ~~5.3.2, 5.3.3, 5.3.5, and 5.3.6~~ show cross sectional views of the HI-STORM 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. ~~These figures~~*Figures 5.3.1 through 5.3.3* were created with the MCNP two-dimensional plotter and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figure 5.3.7 shows a cross sectional view of the 125-ton HI-TRAC with the MPC-24 inside as it was modeled in MCNP. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed.

Figure 5.3.10 shows a cross sectional view of the HI-STORM 100 overpack with the as-modeled thickness of the various materials. *These dimensions are the same for the HI-STORM 100S overpack.* Figure*s* 5.3.11 *and 5.3.18 are* ~~is an~~ axial representation*s* of the HI-STORM 100 *and HI-STORM 100S* overpack*s*, *respectively,* with the various as-modeled dimensions indicated.

Figures 5.3.12 and 5.3.13 show axial cross-sectional views of the 100- and 125-ton HI-TRAC transfer casks, respectively, with the as-modeled dimensions and materials specified. Figures 5.3.14 and 5.3.15 show fully labeled radial cross-sectional views of the 100- and 125-ton HI-

TRAC casks, respectively. Finally, Figures 5.3.16 and 5.3.17 show fully labeled diagrams of the transfer lids for the 100- and 125-ton HI-TRAC casks.

To reduce the gamma dose around the inlet and outlet vents, stainless steel cross plates, designated gamma shield cross plates† (see Figures *5.3.11 and 5.3.18*), have been installed inside all vents. The steel in these plates effectively attenuates the fuel and ^{60}Co gammas that dominated the dose at these locations prior to their installation. *Figure 5.3.19 shows two designs for the gamma shield cross plates to be used in the inlet and outlet vents. The designs in the top portion of the figure are mandatory for use in the HI-STORM 100 and 100S overpacks during normal storage operations and were assumed to be in place in the shielding analysis. The designs in the bottom portion of the figure may be used instead of the mandatory designs in the HI-STORM 100S overpack to further reduce the radiation dose rates at the vents. These optional gamma shield cross plates could further reduce the dose rate at the vent openings by as much as a factor of two.*

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 15 times the pitch and 7 times the pitch, respectively. Homogenization resulted in a noticeable decrease in run time.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The basket material in the top and bottom 0.9 inches where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9-inch area at the top and bottom of the MPC basket.

2. The upper and lower fuel spacers are not modeled, as the fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The fuel spacer length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis, but the fuel spacer materials are not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.

3. For the *MPC-32,* MPC-24, and MPC-68, the MPC basket supports are not modeled. This is conservative since it removes steel that would provide a small

† This design embodiment, formally referred to as "Duct Photon Attenuator," has been disclosed as an invention by Holtec International for consideration by the US Patent Office for issuance of a patent under U.S. law.

increase in shielding. The *optional* aluminum heat conduction elements are also conservatively not modeled.

4. The MPC-24 basket is fabricated from 5/16 inch thick cell plates and 9/32 inch thick angles. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch thick steel. The Boral and sheathing are modeled explicitly. This is conservative since it removes steel that would provide a small amount of additional shielding.

5. In the modeling of the BWR fuel assemblies, the zircaloy flow channels were not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.

6. In the MPC-24, 12 of the 24 Boral panels on the periphery have a reduced width. Conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches.

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models of the HI-STORM 100 and HI-TRAC. Therefore, the models do not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

HI-TRAC Modeling Discrepancies

1. The pocket trunnion on the 125-ton HI-TRAC was modeled as penetrating the lead. This is conservative for gamma dose rates as it reduces shielding thickness.

2. The lifting blocks in the top lid of the 125-ton HI-TRAC were not modeled. Holtite-A was modeled instead. This is a small, localized item and will not impact the dose rates.

3. The door side plates that are in the middle of the transfer lid of the 125-ton HI-TRAC are not modeled. This is acceptable because the dose location calculated on the bottom of the transfer lid is in the center.

4. *The outside diameter of the Holtite-A portion of the top lid of the 125-ton HI-TRAC was modeled as 4 inches larger than it is due to a recent design enhancement. This is acceptable because the peak dose rates on the top lid occur on the inner portions of the lid.*

5. *The lifting trunnion blocks on both HI-TRACs were modeled as 8 inches in height (see Figure 5.3.12 and 5.3.13). Through a recent design enhancement, this*

dimension has increased to 10 inches. The effect of this change is to replace 2 inches of lead by steel. This change does not impact the analysis presented in this chapter since it is a very localized effect and the dose rates reported in that area are averaged over the circumference of the cask.

6. *The region of the water jacket cutout below both lifting trunnions on the 100-ton HI-TRAC (see Figures 5.3.12 and 5.3.13) has been extended circumferentialy by one water channel compared to the analysis. This is acceptable because the design enhancement does not increase the dose rate in that region, it simply increases the width of the region. Therefore, the dose rate calculated in that region and reported in Section 5.4.1 is still valid.*

HI-STORM Modeling Discrepancies

1. The steel channels in the cavity between the MPC and overpack were not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.

2. The bolt anchor blocks were not explicitly modeled. Concrete was used instead. These are small, localized items and will not impact the dose rates.

3. *In the HI-STORM 100S model, the exit vents were modeled as being inline with the inlet vents. In practice, they are rotated 45 degrees and positioned above the short radial plates. Therefore, this modeling change has the exit vents positioned above the full length radial plates. This modeling change has minimal impact on the dose rates at the exit vents.*

4. *The short radial plates in the HI-STORM 100S overpack were modeled in MCNP even though they are optional.*

5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.8 and 5.3.9 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM 100 System. The axial locations of the Boral, basket, inlet vents, and outlet vents are shown in these figures.

5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. The gamma shield cross plates located in the inlet and outlet vents, which effectively reduce the gamma dose in these locations, are modeled explicitly.

The MCNP model of the HI-TRAC transfer cask describes the lifting trunnions, pocket trunnions, and the opening in the HI-TRAC top lid. The ribs through the HI-TRAC water jacket are also modeled. Streaming considerations through these trunnions and fins are discussed in Section 5.4.1.

The design of the HI-STORM 100 System, as described in the Design Drawings in Chapter 1, has eliminated all other possible streaming paths. Therefore, the MCNP model does not represent any additional streaming paths. A brief justification of this assumption is provided for each penetration.

- The lifting trunnions will remain installed in the HI-TRAC transfer cask. No credit is taken for any part of the trunnion that extends from the HI-TRAC body.

- The pocket trunnions of the HI-TRAC are modeled as solid blocks of steel. No credit is taken for any part of the pocket trunnion that extends beyond the water jacket.

- The threaded holes in the MPC lid are plugged with solid plugs during storage and, therefore, do not create a void in the MPC lid.

- The drain and vent ports in the MPC lid are designed to eliminate streaming paths. *The holes in the vent and drain port cover plates are filled with a set screw and plug weld.* The steel lost in the MPC lid at the port location is replaced with a block of steel approximately 6 inches thick located directly below the port opening and attached to the underside of the lid. This design feature is shown on the Design Drawings in Chapter 1. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid plate.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM 100 System and HI-TRAC shielding analyses are given in Tables 5.3.2 and 5.3.3. All of the materials and their actual geometries are represented in the MCNP model.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask is equipped with a water jacket providing radial neutron shielding. Demineralized water will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

Since the HI-STORM 100S does not have the inner shield shell present, the minimum density of the concrete in the body (not the lid or pedestal) of the overpack has been increased slightly to compensate for the change in shielding relative to the HI-STORM 100 overpack. Table 5.3.2 shows the concrete composition and densities that were used for the HI-STORM 100 and HI-STORM 100S overpacks. Since the density of concrete is increased by altering the aggregate that is used, the composition of the slightly denser concrete was calculated by keeping the same mass of water as the 2.35 gm/cc composition and increasing all other components by the same ratio.

Sections 4.4 and 4.5 demonstrate that all materials used in the HI-STORM and HI-TRAC remain below their design temperatures as specified in Table 2.2.3 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Section 5.1.2, there is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield (water) in the HI-TRAC as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR					
Lower End Fitting	0.0	7.375	7.375	SS304	SS304
Space	7.375	8.375	1.0	zircaloy	void
Fuel	8.375	152.375	144	fuel & zircaloy	fuel
Gas Plenum Springs	152.375	156.1875	3.8125	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875	160.5625	4.375	SS304 & zircaloy	SS304
Upper End Fitting	160.5625	165.625	5.0625	SS304	SS304
BWR					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144	fuel & zircaloy	fuel
Space	151.385	157.385	6	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the lower fuel spacer must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Uranium Oxide	10.412	^{235}U	2.9971(BWR) 3.2615(PWR)
		^{238}U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral†	2.644	^{10}B	4.4226 (MPC-68 *and MPC-32* in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 4.367 (MPC-24 in HI-TRAC)
		^{11}B	20.1474 (MPC-68 *and MPC-32* in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 19.893 (MPC-24 in HI-TRAC)
		Al	68.61 (MPC-68 *and MPC-32* in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 69.01 (MPC-24 in HI-TRAC)
		C	6.82 (MPC-68 *and MPC-32* in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 6.73 (MPC-24 in HI-TRAC)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

† All B-10 loadings in the Boral compositions are conservatively lower than the values defined in the Bill of Materials.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Neutron Shield Holtite-A	1.61	C	27.66039
		H	5.92
		Al	21.285
		N	1.98
		O	42.372
		^{10}B	0.14087
		^{11}B	0.64174
BWR Fuel Region Mixture	4.29251	^{235}U	2.4966
		^{238}U	70.9315
		O	9.8709
		Zr	16.4046
		N	8.35E-05
		Cr	0.0167
		Fe	0.0209
		Sn	0.2505
PWR Fuel Region Mixture	3.869939	^{235}U	2.7652
		^{238}U	71.9715
		O	10.0469
		Zr	14.9015
		Cr	0.0198
		Fe	0.0365
		Sn	0.2587

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Lower End Fitting (PWR)	1.0783	SS304	100
Gas Plenum Springs (PWR)	0.1591	SS304	100
Gas Plenum Spacer (PWR)	0.1591	SS304	100
Upper End Fitting (PWR)	1.5410	SS304	100
Lower End Fitting (BWR)	1.4862	SS304	100
Gas Plenum Springs (BWR)	0.2653	SS304	100
Expansion Springs (BWR)	0.6775	SS304	100
Upper End Fitting (BWR)	1.3692	SS304	100
Handle (BWR)	0.2572	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.9140 (water jacket)	H	11.2
	0.9619 (inside MPC)	O	88.8

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Concrete	2.35 *HI-STORM 100 and lid and pedestal of HI-STORM 100S*	H	0.6
		O	50.0
		Si	31.5
		Al	4.8
		Na	1.7
		Ca	8.3
		Fe	1.2
		K	1.9
Concrete	*2.48* *HI-STORM 100S body*	*H*	*0.569*
		O	*49.884*
		Si	*31.594*
		Al	*4.814*
		Na	*1.705*
		Ca	*8.325*
		Fe	*1.204*
		K	*1.905*

Table 5.3.3

COMPOSITION OF THE FUEL PELLETS IN THE MIXED OXIDE FUEL ASSEMBLIES

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Mixed Oxide Pellets	10.412	^{238}U	85.498
		^{235}U	0.612
		^{238}Pu	0.421
		^{239}Pu	1.455
		^{240}Pu	0.034
		^{241}Pu	0.123
		^{242}Pu	0.007
		O	11.85
Uranium Oxide Pellets	10.412	^{238}U	86.175
		^{235}U	1.975
		O	11.85

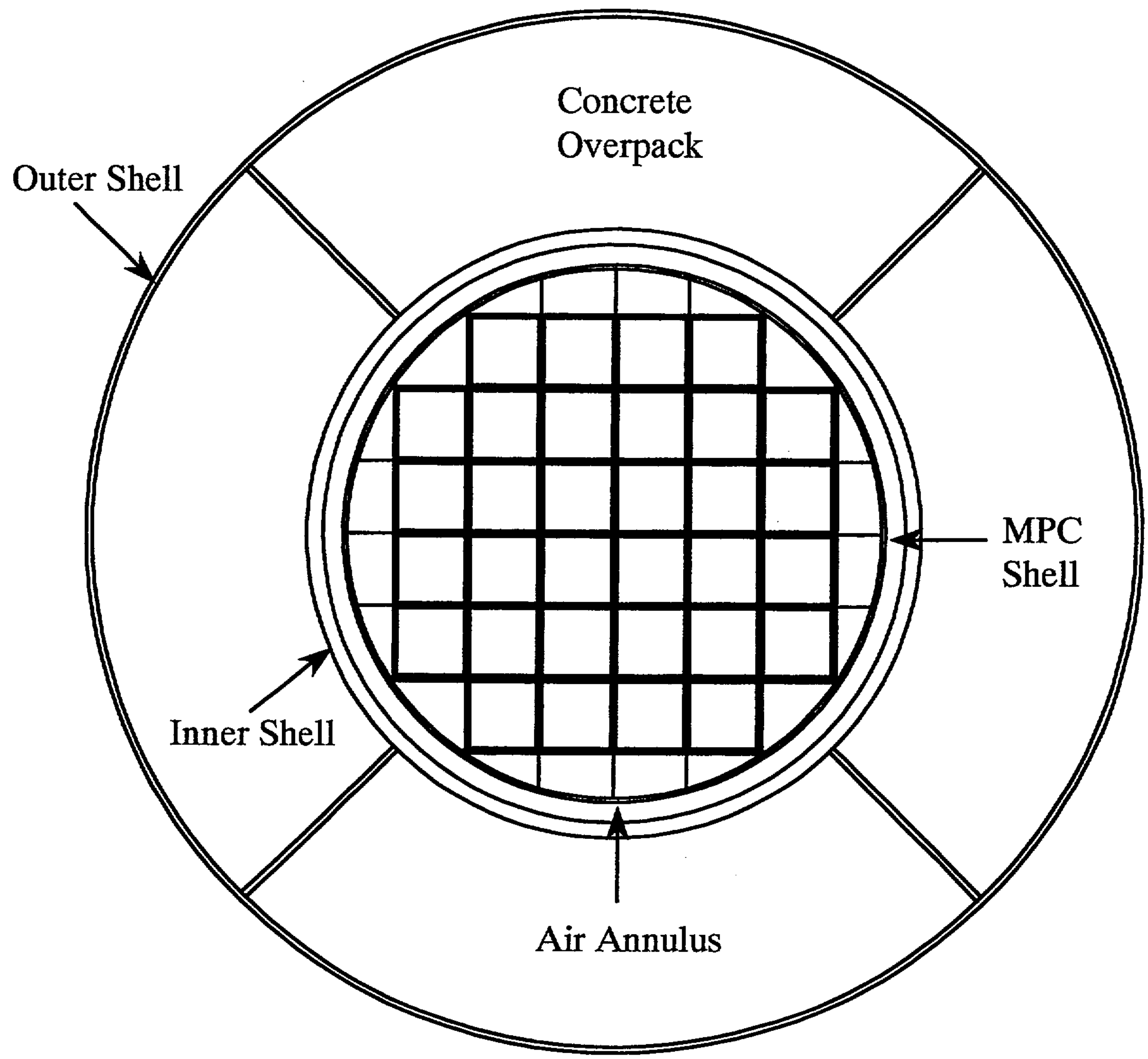


FIGURE 5.3.1; HI-STORM 100 OVERPACK WITH MPC-32 CROSS SECTIONAL VIEW AS MODELLED IN MCNP†

† This figure is drawn to scale using the MCNP plotter.

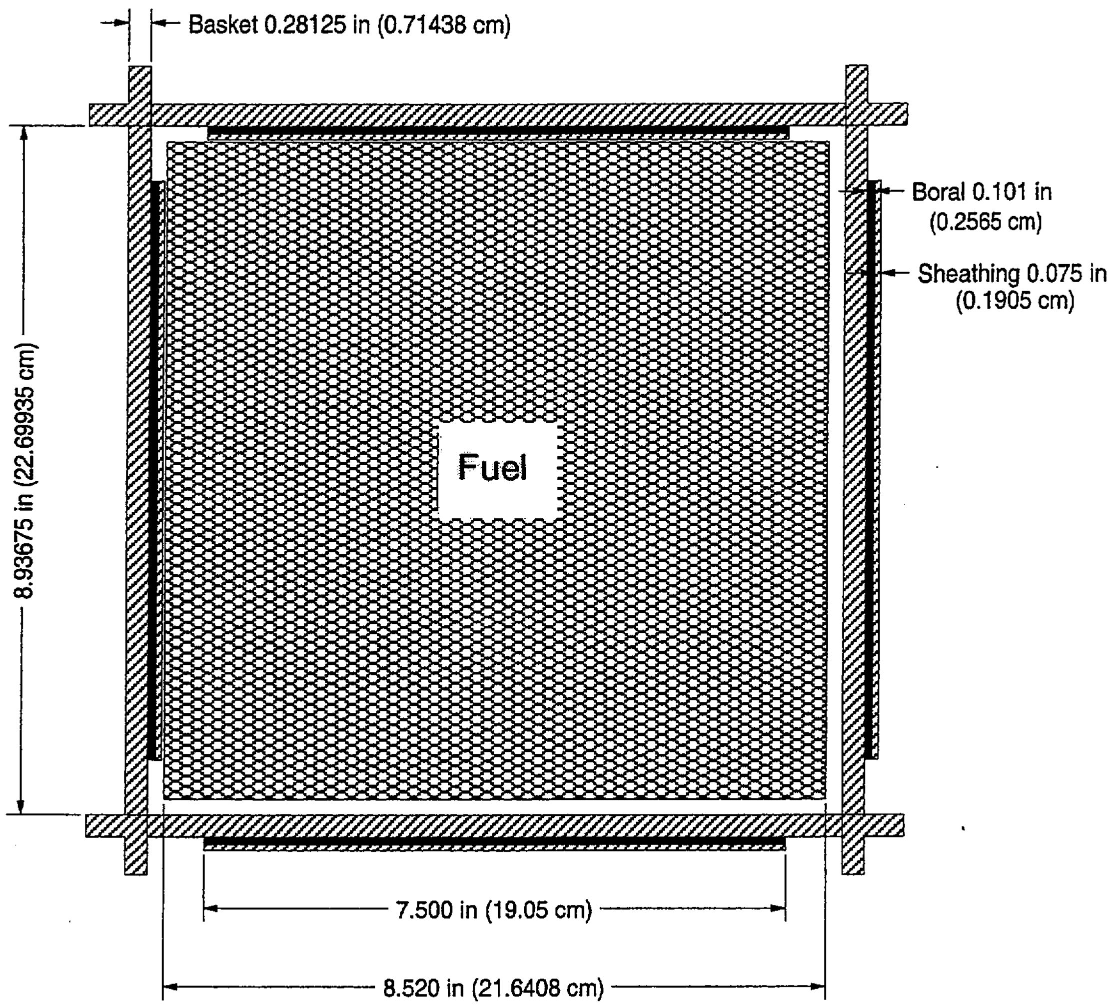


FIGURE 5.3.4; CROSS SECTIONAL VIEW OF AN MPC-32 BASKET CELL AS MODELED IN MCNP

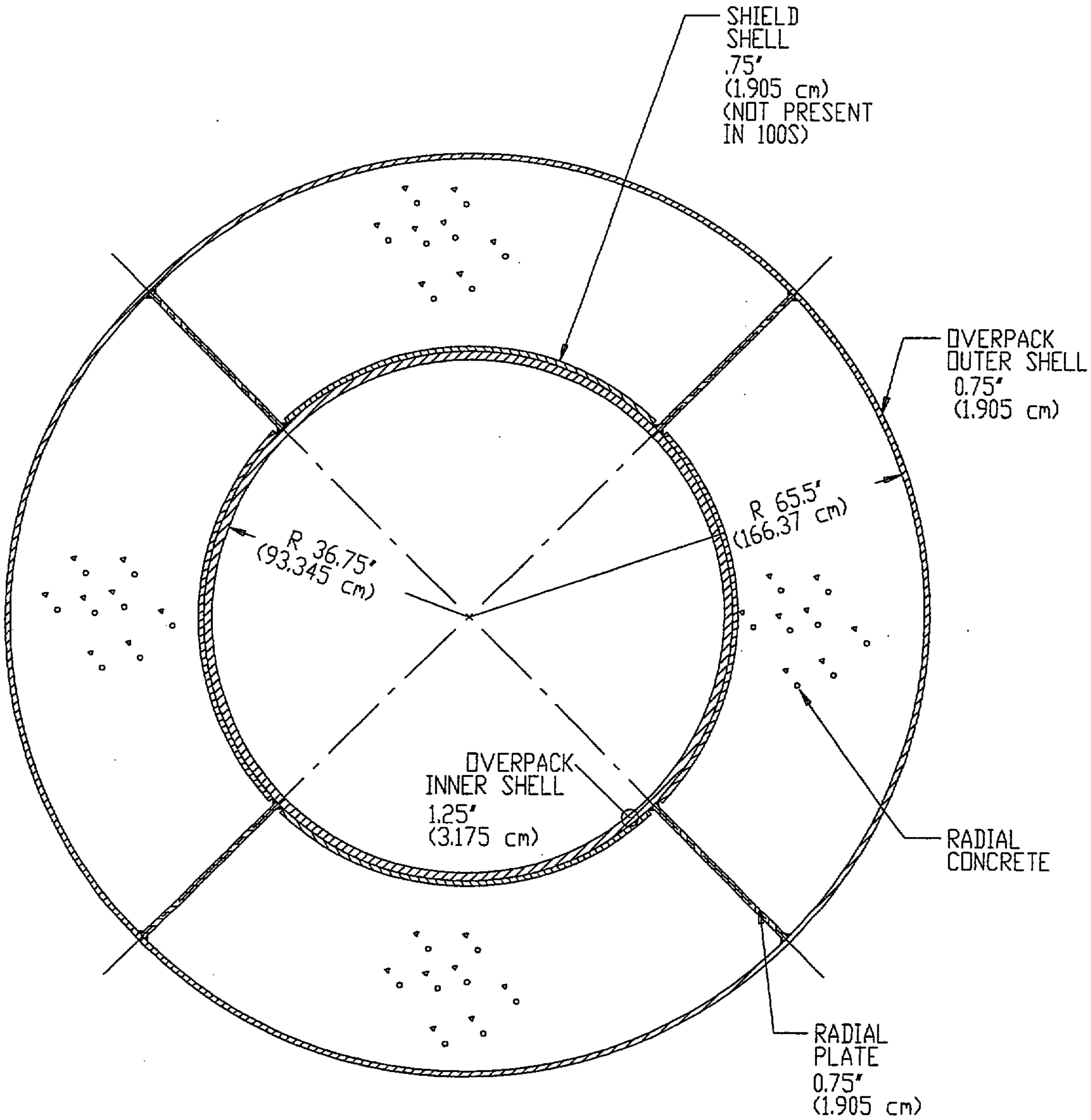


FIGURE 5.3.10; CROSS SECTION OF HI-STORM 100 OVERPACK

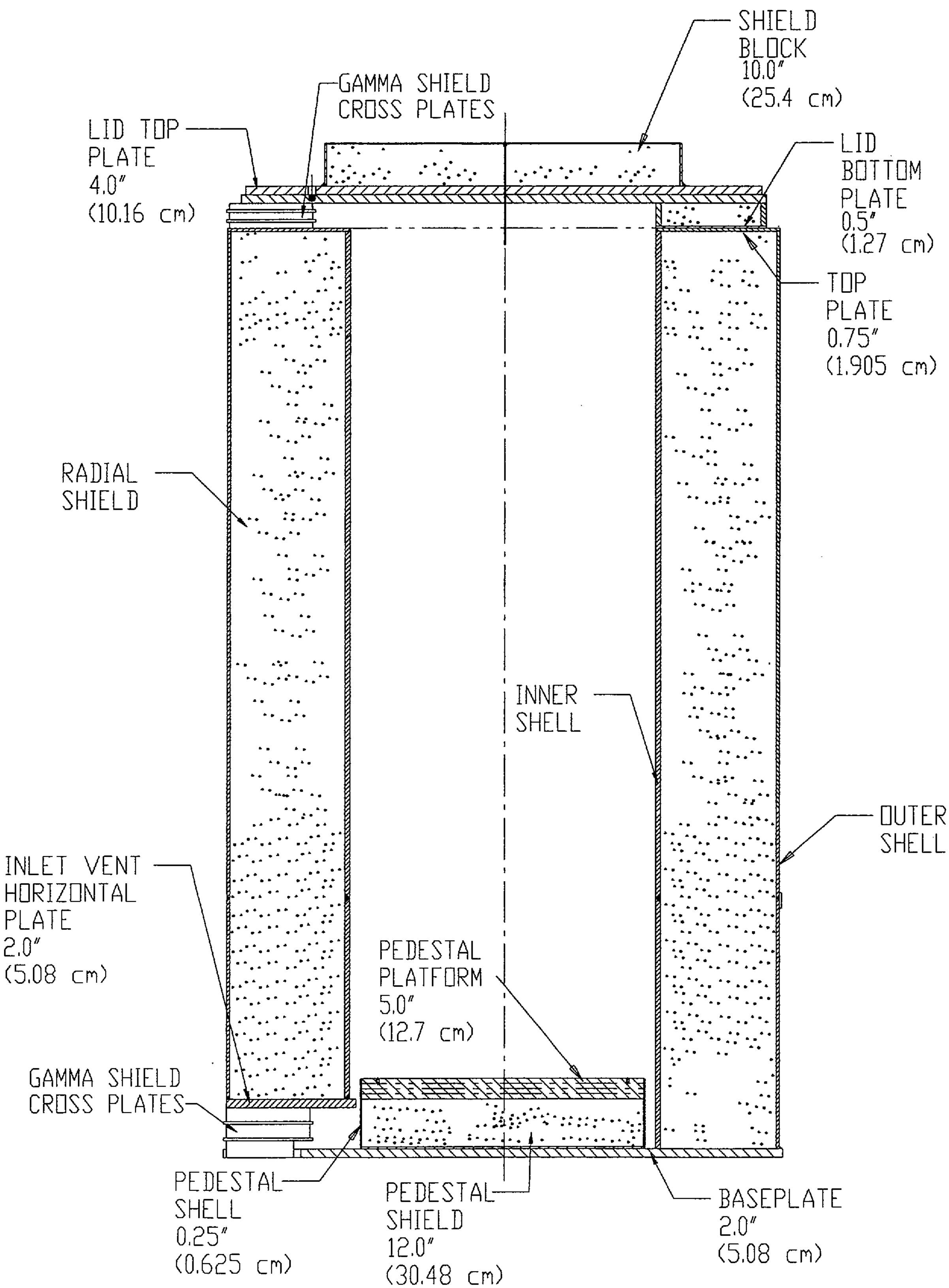
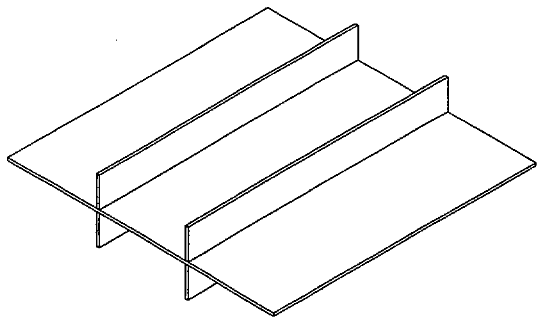
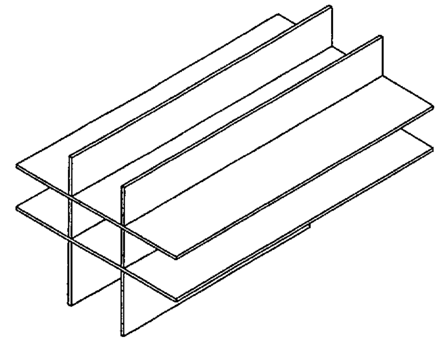


FIGURE 5.3.18; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

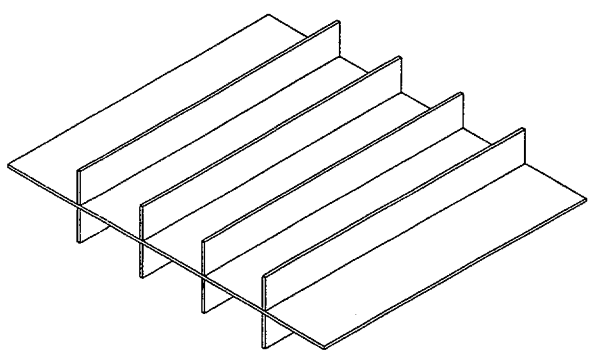


OUTLET VENT

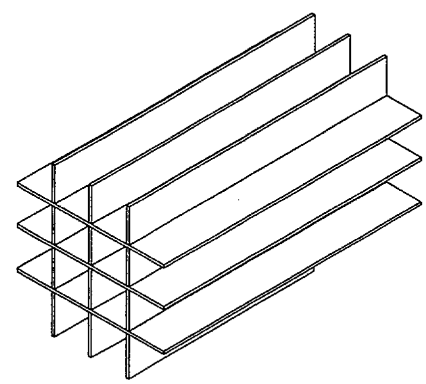


INLET VENT

MANDATORY GAMMA SHIELD CROSS PLATES FOR HI–STORM 100
AND HI–STORM 100S



OUTLET VENT



INLET VENT

OPTIONAL GAMMA SHIELD CROSS PLATES FOR HI–STORM 100S
THAT MAY BE USED INSTEAD OF THE MANDATORY DEVICES.

FIGURE 5.3.19: GAMMA SHIELD CROSS PLATE CONFIGURATION OF HI–STORM 100 AND HI–STORM 100S

5.4 SHIELDING EVALUATION

The MCNP-4A code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B-V data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.11 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.11 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.C. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

The HI-STORM shielding analysis was performed for conservative burnup and cooling time combinations which bound the uniform and regionalized loading specifications for zircaloy clad fuel specified in Appendix B to the CoC. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68. ~~Technical Specification Table 2.1-4, Appendix 12.A, requires that, in the MPC-24, for a minimum cooling~~

~~time of 5 years, the maximum burnup is 31,300 MWD/MTU, and for 15 year cooling the maximum burnup is 44,700 MWD/MTU. Since the burnup and cooling times analyzed for the HI-STORM overpack with the MPC-24 were 45,000 MWD/MTU and 5 year cooling, the shielding analysis presented is conservatively bounding for the MPC-24 in HI-STORM.~~

~~Technical Specification Table 2.1-4, Appendix 12.A, requires that, in the MPC-68, for a minimum cooling time of 5 years, the maximum burnup is 29,900 MWD/MTU, and for 15 year cooling the maximum burnup is 41,700 MWD/MTU. Since the burnup and cooling times analyzed for the HI-STORM overpack with the MPC-68 were 45,000 MWD/MTU and 5 year cooling, the shielding analysis presented is conservatively bounding for the MPC-68 in HI-STORM.~~

Tables 5.1.*1*~~2~~ ~~and~~ *through* 5.1.3 provide the maximum dose rates adjacent to the HI-STORM overpack during normal conditions for each of the MPCs. Tables 5.1.*4*~~5~~ ~~and~~ *through* 5.1.6 provide the maximum dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Sections 5.1.1 and 5.1.2.

Tables 5.1.7 and 5.1.8 provide dose rates for the 100-ton and 125-ton HI-TRAC transfer casks, *respectively,* with the MPC-24 loaded with design basis fuel in the normal condition, in which the MPC is dry and the HI-TRAC water jacket is filled with water. Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with the water jacket filled with water (condition in which welding operations are performed). Dose locations 4 and 5, which are on the top and bottom of the HI-TRAC were not calculated at the one-meter distance for these configurations. For the conditions involving a fully flooded MPC, the internal water level was 10 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table~~s~~ 5.1.7 ~~and 5.1.8~~ indicates that the dose rates in the upper and lower portions of the HI-TRAC are reduced by about 50% with the water in the MPC. The dose at the center of the HI-TRAC is reduced by approximately 50% when there is also water in the water jacket and is essentially unchanged when there is no water in the water jacket as compared to the normal condition results shown in Table~~s~~ 5.1.7 ~~and 5.1.8~~.

The burnup and cooling time combination of ~~35,000~~*42,500* MWD/MTU and 5 years was selected for the 100-ton *MPC-24* HI-TRAC analysis because this combination of burnup and cooling time results in the highest dose rates, and therefore, bounds all other requested combinations in the 100-ton HI-TRAC. For comparison, dose rates corresponding to a burnup of ~~45,000~~*52,500* MWD/MTU and ~~9~~*10* year cooling time for the MPC-24 are provided in Table 5.4.4. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results clearly indicate that as the burnup and cooling time increase, the reduction in the gamma dose rate due to the increased cooling time results in a net decrease in the total dose rate. This result is due to the fact that the dose rates surrounding the 100-ton HI-TRAC transfer cask are gamma dominated.

In contrast, the dose rates surrounding the 125-ton HI-TRAC transfer cask have significantly higher neutron component. Therefore, the dose rates at ~~45,000~~*57,500* MWD/MTU burnup and ~~9~~ *12* year cooling are slightly higher than the dose rates at ~~35,000~~*42,500* MWD/MTU burnup and 5 year cooling. The dose rates for the 125-ton HI-TRAC with the MPC-24 at ~~45,000~~*57,500* MWD/MTU and ~~9~~*12* year cooling are listed in Table~~s~~ ~~5.1.7 and~~ 5.1.8 of Section 5.1. For comparison, dose rates corresponding to a burnup of ~~35,000~~*42,500* MWD/MTU and 5 year cooling time for the MPC-24 are provided in Table 5.4.5.

Tables 5.4.9 and 5.4.10 provide~~d~~ dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-68 at burnup and cooling time combinations of ~~30,000~~*40,000* MWD/MTU and 5 years and ~~45,000~~*50,000* MWD/MTU and ~~12~~ *10* years, respectively. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-68 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-68 at the center of the HI-TRAC, on-contact, and at locations 1 to 2 feet away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

Tables 5.4.11 and 5.4.12 provide dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-32 at burnup and cooling time combinations of 32,500 MWD/MTU and 5 years and 45,000 MWD/MTU and 10 years, respectively. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-32 case than in the MPC-24 case. However, the MPC-24 produces comparable or higher dose rates than the MPC-32 at the center of the HI-TRAC, on-contact, and at locations 1 to 2 feet away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

As mentioned in Section 5.0, all MPCs offer a regionalized loading pattern as described in Appendix B to the CoC. This loading pattern authorizes fuel of higher decay heat than uniform loading (i.e. higher burnups and shorter cooling times) to be stored in the center region, region 1, of the MPC. The outer region, region 2, of the MPC in regionalized loading is authorized to store fuel of lower decay heat than uniform loading (i.e. lower burnups and longer cooling times). From a shielding perspective, the older fuel on the outside provides shielding for the inner fuel in the radial direction. Regionalized patterns were specifically analyzed in each MPC in the 100-ton HI-TRAC. Based on analysis using the same burnup and cooling times in region 1 and 2 the following percentages were calculated for dose location 2 on the 100-ton HI-TRAC.

- *Approximately 21%, 27%, and 8% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24 respectively. Region 1 contains 12 (38% of total), 32 (47% of total), and 4 (17% of total) assemblies in the MPC-32, MPC-68, and MPC-24 respectively.*
- *Approximately 1%, 2%, and 0.2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24 respectively.*

These results clearly indicate that the outer fuel assemblies shield almost all of the gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. In the axial direction, regionalized loading results in higher dose rates in the center portion of the cask since the region 2 assemblies are not shielding the region 1 assemblies for axial dose locations.

All burnup and cooling time combinations for regionalized loading were analyzed and compared to the dose rates from uniform loading patterns. It was concluded that, in general, the radial dose rates from regionalized loading are bounded by the radial dose rates from uniform loading patterns. Therefore, dose rates for specific regionalized loading patterns are not presented in this chapter. In the axial direction, the reverse may be true since the inner fuel assemblies in a regionalized loading pattern have a higher burnup than the assemblies in the uniform loading patterns. However, as depicted in the graphical data in Section 5.1.1, the dose rate along the pool or transfer lids decrease substantially moving radially outward from the center of the lid. Therefore, this increase in the dose rate in the center of the lids due to regionalized loading does not significantly impact the occupational exposure. Section 5.4.9 provides additional discussion on regionalized loading dose rates compared to uniform loading dose rates.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 5% and the relative error for the individual dose components was typically less than 10%.

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions and Azimuthal Variations

The HI-STORM 100 overpack and the HI-TRAC utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would result in a localized reduction in the photon dose. In addition to the fins, the pocket trunnions in the HI-TRAC are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin.

Since the HI-STAR 100 System utilizes fins and pocket trunnions similar to the HI-TRAC, the streaming analysis performed for HI-STAR 100 is applicable to this discussion. The reader is referred to the HI-STAR 100 applications under Docket Nos. 71-9261 and 72-1008 for the discussion on streaming.

The general conclusion from the HI-STAR analysis was that a streaming effect through the fins does exist and is detectable on contact to the overpack's surface. However, at a distance of one-meter from the surface, the streaming is no longer detectable. Streaming through pocket trunnions is more noticeable; however, the increase in dose is only a factor of 1.3 at contact and also is not significant at one-meter distance.

These conclusions indicate that the results presented in this section are unaffected by streaming through the fins or pocket trunnions.

Below each lifting trunnion, there is a localized area where the water jacket has been reduced in height by 4.125 inches to accommodate the lift yoke (see Figures 5.3.12 and 5.3.13). This area experiences a significantly higher than average dose rate on contact of the HI-TRAC. The peak dose in this location is *2.0 Rem/hr for the MPC-32,* ~~1.5~~*1.9* Rem/hr for the MPC-68 and ~~1.1~~*1.8* Rem/hr for the MPC-24 in the 100-ton HI-TRAC and ~~300~~*516* mrem/hr for the MPC-24 in the 125-ton HI-TRAC. At a distance of 1 to 2 feet from the edge of the HI-TRAC the localized effect is greatly reduced. This dose rate is acceptable because during lifting operations the lift yoke will be in place, which, due to the additional lift yoke steel (~3 inches), will greatly reduce the dose rate. However, more importantly, people will be prohibited from being in the vicinity of the lifting trunnions during lifting operations as a standard rigging practice. In addition the lift yoke is remote in its attachment and detachment, further minimizing personnel exposure. Immediately following the detachment of the lift yoke, in preparation for closure operations, temporary shielding ~~will~~*may* be placed in this area. Any temporary shielding (e.g., lead bricks, water tanks, lead blankets, steel plates, etc.) is sufficient to attenuate the localized hot spot. The operating procedure in Chapter 8 ~~specify~~*discusses* the placement of temporary shielding in this area. For the 100-ton HI-TRAC, the ~~mandated~~*optional* temporary shield~~ing~~ *ring* will replace the water that was lost from the axial reduction in the water jacket thereby eliminating the localized hot spot. When the HI-TRAC is in the horizontal position, during transport operations, it will (at a minimum) be positioned a few feet off the ground by the transport vehicle and therefore this location below the lifting trunnions will be positioned above people which will minimize the effect on personnel exposure. In addition, good operating practice will dictate that personnel remain at least a few feet away from the transport vehicle. During vertical transport of a loaded HI-TRAC, the localized hot spot will be even further from the operating personnel. Based on these considerations, the conclusion is that this localized hot spot does not significantly impact the personnel exposure.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

5.4.2.1 Dresden 1 and Humboldt Bay Damaged Fuel

As discussed in Section 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of storing intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies.

For the damaged fuel and fuel debris accident condition, it is conservatively assumed that the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the Design Drawings of Chapter 1, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid. Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches.

Dividing the total fuel gamma source for ~~damaged~~ *a 6x6* fuel *assembly* in Table 5.2.7 by the 80 inch rubble height provides a gamma source per inch of 3.41E+12 photon/s. Dividing the total neutron source for ~~damaged~~ *a 6x6* fuel *assembly* in Table 5.2.18 by 80 inches provides a neutron source per inch of 2.75E+05 neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.~~23E~~*08E*+13 photon/s and ~~1.33E+06~~*9.17E+05* neutron/s, respectively. These BWR design basis values were calculated by dividing the total source strengths for ~~45~~*40*,000 MWD/MTU and 5 year cooling in Tables 5.2.6 and 5.2.17 by the active fuel length of 144 inches. Therefore, ~~the design basis~~ damaged *Dresden 1 and Humboldt Bay* fuel *assemblies are* ~~assembly is~~ bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates *from Dresden 1 or Humboldt Bay fuel assemblies* are provided as they are bounded by the intact fuel analysis.

5.4.2.2 Generic PWR and BWR Damaged Fuel

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact. As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was performed to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Only the 100-ton HI-TRAC was analyzed because it can be concluded that if the dose rate change is not significant for the 100-ton HI-TRAC then the change will not be significant for the 125-ton HI-TRAC or the HI-STORM overpacks.

Fuel debris or a damaged fuel assembly which has collapsed can have an average fuel density which is higher than the fuel density for an intact fuel assembly. If the damaged fuel assembly were to fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. This scenario was analyzed with MCNP-4A in a conservative bounding fashion to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 (4 peripheral locations in the MPC-24E or MPC-24EF) and the MPC-68 (16 peripheral locations) with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

Tables 5.4.13 and 5.4.14 provide the results for the MPC-24 and MPC-68, respectively. Only the radial dose rates are provided since the axial dose rates will not be significantly affected because the damaged fuel assemblies are located on the periphery of the baskets. A comparison of these results to the results in Tables 5.1.7 and 5.4.9 indicate that the dose rates in the top and bottom portion of the 100-ton HI-TRAC increase by less than 20% while the dose rate in the center of the HI-TRAC actually decreases a little bit. The increase in the bottom and top is due to the assumed flat power distribution. The dose rates shown in Tables 5.4.13 and 5.4.14 were averaged over the circumference of the cask. Since almost all of the peripheral cells in the MPC-68 are filled with DFCs, an azimuthal variation would not be expected for the MPC-68. However, since there are only 4 DFCs in the MPC-24E, an azimuthal variation in dose due to the damaged fuel/fuel debris might be expected. Therefore, the dose rates were evaluated in four smaller regions, one outside each DFC, that encompass about 44% of the circumference. There was no significant change in the dose rate as a result of the localized dose calculation. These results indicate that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter.

5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR Chapter 12 assigns ultimate compliance responsibilities to the site licensee. Therefore, this subsection describes, by example, the general methodology for performing site boundary dose calculations. The site-specific fuel characteristics, burnup, cooling time, and the site characteristics would be factored into the evaluation performed by the licensee.

As an example of the methodology, the dose from a single HI-STORM overpack loaded with an MPC-24 and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters were evaluated with MCNP. In the model, the casks were placed on an infinite slab of dirt to account for earth-shine effects. The atmosphere was represented by dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 700 meters. This is more than sufficient to properly account for skyshine effects. The models included either 500 or 1050 meters of air around the cask. Based on the behavior of the dose rate as a function of distance, 50 meters of air, beyond the detector locations, is sufficient to account for back-scattering. Therefore, the HI-STORM MCNP off-site dose models account for back scattering by including more than 50 meters of air beyond the detector locations for all cited dose rates. Since gamma back-scattering has an effect on the off-site dose, it is recommended that the site-specific evaluation under 10CFR72.212 include at least 50 to 100 meters of air, beyond the detector locations, in the calculational models.

The MCNP calculations of the off-site dose used a two-stage process. In the first stage a binary surface source file (MCNP terminology) containing particle track information was written for particles crossing the outer radial and top surfaces of the HI-STORM overpack. In the second stage of the calculation, this surface source file was used with the particle tracks originating on the outer edge of the overpack and the dose rate was calculated at the desired location (hundreds of meters away from the overpack). The results from this two-stage process are statistically the same as the results from a single calculation. However, the advantage of the two-stage process is that each stage can be optimized independently.

The annual dose, assuming 100% occupancy (8760 hours), at 200 meters from one cask is presented in Table 5.4.6 for the design basis ~~maximum~~ burnup and ~~minimum~~ cooling times analyzed. This table indicates that the dose due to neutrons is ~~5~~ 7 % of the total dose. This is an important observation because it implies that simplistic analytical methods such as point kernel techniques may not properly account for the neutron transmissions and could lead to low estimates of the site boundary dose.

The annual dose, assuming 8760 hour occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of the HI-STORM 100 overpack was calculated at the distance desired. Dose value = A.

2. The annual dose from the radiation leaving the top of the HI-STORM 100 overpack was calculated at the distance desired. Dose value = B.

3. The annual dose from the radiation leaving the side of a HI-STORM 100 overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7 for the bounding burnup and cooling time of ~~45,000~~*52,500* MWD/MTU and 5-year cooling. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM 100 overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

As an example, the dose from a 2x3 array at 300 meters is presented.

1. The annual dose from the side of a single cask: Dose A = *5.20*~~4.40~~
2. The annual dose from the top of a single cask: Dose B = *6.57*~~4.23~~e-2
3. The annual dose from the side of a cask positioned behind another cask:
   Dose C = *1.04*~~0.88~~

Using the formula shown above (Z=3), the total dose at 300 meters from a 2x3 array of HI-STORM overpacks is ~~16.09~~*19.11* mrem/year, assuming a 8760 hour occupancy.

An important point to notice here is that the dose from the side of the back row of casks is 16 % of the total dose. This is a significant contribution and one that would probably not be accounted for properly by simpler methods of analysis.

The results for various typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis. Therefore, a site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

5.4.4 Stainless Steel Clad Fuel Evaluation

Table 5.4.8 presents the dose rates at the center of the HI-STORM 100 overpack, adjacent and at one meter distance, from the stainless steel clad fuel. These dose rates, when compared to Tables ~~5.1.2, 5.1.3, 5.1.5, and~~*5.1.1 through* 5.1.6, are similar to the dose rates from the design basis zircaloy clad fuel, indicating that these fuel assemblies are acceptable for storage.

As described in Section 5.2.3, it would be incorrect to compare the total source strength from the stainless steel clad fuel assemblies to the source strength from the design basis zircaloy clad fuel assemblies since these assemblies do not have the same active fuel length and since there is a significant gamma source from Cobalt-60 activation in the stainless steel. Therefore it is necessary to calculate the dose rates from the stainless steel clad fuel and compare them to the dose rates from the zircaloy clad fuel. In calculating the dose rates, the source term for the stainless steel fuel was calculated with an artificial active fuel length of 144 inches to permit a simple comparison of dose rates from stainless steel clad fuel and zircaloy clad fuel at the center of the HI-STORM 100 overpack. Since the true active fuel length is shorter than 144 inches and since the end fitting masses of the stainless steel clad fuel are assumed to be identical to the end fitting masses of the zircaloy clad fuel, the dose rates at the other locations on the overpack are bounded by the dose rates from the design basis zircaloy clad fuel, and therefore, no additional dose rates are presented.

5.4.5 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the source terms for the BWR design basis zircaloy clad fuel assembly (GE 7x7) which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Dividing the total fuel gamma source for the MOX fuel in Table 5.2.22 by the 110 inch active fuel height provides a gamma source per inch of 2.36e+12 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.23 by 110 inches provides a neutron source strength per inch of 3.06e+5 neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.~~23e~~*08e*+13 photons/s and ~~1.33e+6~~*9.17e+5* neutrons/s. These BWR design basis values were calculated by dividing the total source strengths for ~~45~~*40*,000 MWD/MTU and 5 year cooling in Tables 5.2.6 and 5.2.17 by the active fuel length of 144 inches. This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Section 5.4.2.*1*, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches. The resulting gamma and neutron source strengths are 3.25e+12 photons/s and 4.21e+5 neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post accident configuration.

5.4.6 Non-Fuel Hardware and Control Components

As discussed in Section 5.2.4, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM 100 System. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. BPRAs and TPDs are authorized for unrestricted storage in an MPC while the CRAs and APSRs are restricted to the center four locations in the MPC-24, MPC-24E, MPC-24EF and MPC-32. The calculation of the source term and a description of the bounding fuel devices was provided in Section 5.2.4. The dose rate due to BPRAs and TPDs being stored in a fuel assembly was explicitly calculated. Table 5.4.15 provides the dose rates at various locations on the surface and one meter from the 100-ton HI-TRAC due to the BPRAs and TPDs for the MPC-24 and MPC-32. These results were added to the totals in the other table to provide the total dose rate with BPRAs. Table 5.4.15 indicates that the dose rates from BPRAs bound the dose rates from TPDs.

As discussed in Section 5.2.4, two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs. The dose rate due to CRAs and APSRs being stored in the inner four fuel locations was explicitly calculated for dose locations around the 100-ton HI-TRAC. Tables 5.4.16 and 5.4.17 provide the results for the different configurations of CRAs and APSRs, respectively, in the MPC-24 and MPC-32. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is minimal and the dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, as noted in Tables 5.4.16 and 5.4.17, the dose rate at the edge of the transfer lid is almost negligible due to APSRs and CRAs. Therefore, even though the dose rates calculated (using a very conservative source term evaluation) are daunting, they do not pose a risk from an operations perspective because they are localized in nature. Section 5.1.1 provides additional discussion on the acceptability of the relatively high localized doses on the bottom of the HI-TRACs.

5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches in length. The steel rod is approximately 95 inches in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron *source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies). The source strength is approximately 1E+8 neutrons/second."*

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as 5.8E-6 neutrons per gamma (1E+8/865/3.7e+10/0.54) with energy greater than 1.666 MeV or 1.16E+5 neutrons/curie (1E+8/865) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in.). The length of the source is 77.25 inches. The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the beryllium was calculated to be 0.24 inches. The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

*The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch long neutron source was calculated to be 1.04E+8 gammas/sec which would produce a neutron source of 603.2 neutrons/sec (1.04E+8 * 5.8E-6). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies. This activity level would produce a neutron source of 4.63E+6 neutrons/sec (39.9 * 1.16E+5) or 6.0E+4 neutrons/sec/inch (4.63E+6/77.25). These calculations conservatively neglect the reduction in antimony and* *beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.*

Since this is a localized source (77.25 inches in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.18, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

As stated above, the Sb-Be source is encased in a steel rod. Therefore, the gamma source from the activation of the steel was considered assuming a burnup of 120,000 MWD/MTU which is the maximum burnup assuming the Sb-Be source was in the reactor for the entire 18 year life of Dresden Unit 1. The cooling time assumed was 18 years which is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel was bounded by the design basis fuel assembly. In conclusion, storage of a Dresden Unit 1 Sb-Be neutron source in a Dresden Unit 1 fuel assembly is acceptable and bounded by the current analysis.

5.4.8 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.37 and 5.2.7 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.18, bounds the thoria rod neutron spectra, Table 5.2.38, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the 100-ton HI-TRAC and the HI-STORM overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher for the 100-ton HI-TRAC and only 25% lower for the HI-STORM overpack than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.

5.4.9 Regionalized Loading Dose Rate Evaluation

Dose rates were calculated for regionalized loading patterns for the MPC-24, MPC-32, and MPC-68 using MCNP-4A. All burnup and cooling time combinations in Appendix B to the CoC were analyzed for both uniform and regionalized loading. The dose rates for all dose locations reported in this chapter were compared for the uniform loading patterns and the regionalized loading patterns.

It was determined that for the MPC-32, all radial surface and 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated surface dose rates in the axial locations for regionalized loading were less than 15% higher than the uniform dose rates reported in this chapter for the surface of the overpack. At one-meter from the overpack, dose location 4 (in the center) was the only dose location which produced a slightly higher (5%) dose rate for regionalized loading compared to uniform loading.

For the MPC-24 it was determined that the maximum calculated dose rates in the axial direction for regionalized loading were less than 21% higher than the maximum calculated dose rates for uniform loading reported in this chapter. At one meter distance, the uniform loading dose rates reported in this chapter bound the regionalized loading dose rates. In the radial direction, the uniform loading dose rates reported in this chapter bound the regionalized loading dose rates for both surface and one-meter locations.

For the MPC-68 it was determined that all radial surface and 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated surface dose rates in the axial locations for regionalized loading were less than 21% higher than the uniform dose rates reported in this chapter for the surface of the overpack. At one-meter from the overpack, dose locations 4 (in the center) and 5 (transfer lid center) were the only dose locations which produced a slightly higher (5% and 1.5% respectively) dose rate for regionalized loading compared to uniform loading.

Based on these results it can be stated that regionalized loading patterns will reduce the dose rate in the radial direction by shielding the hotter fuel on the inside of the cask with colder fuel on the outside of the cask. However, in the axial direction the localized dose rates in the center of the cask may increase as a result of the regionalized loading pattern. This is a localized effect, which has dissipated at the edge of the cask, and therefore will not result in a significant increase to the occupational exposure rates. In addition, it should be mentioned that the localized increase on the bottom center of the overpack is an area where workers will normally not be present and the increase in the top center of the overpack is an area where workers minimize their stay.

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued)

FLUX-TO-DOSE CONVERSION FACTORS
(FROM [5.4.1])

Neutron Energy (MeV)	Quality Factor	(rem/hr)†/(n/cm^2-s)
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~35,000~~*42,500* MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	*Totals with BPRAs (mrem/hr)*
ADJACENT TO THE 100-TON HI-TRAC					
1	*12.93*	*241.58*	*22.37*	*276.89*	*279.53*
2	*813.20*	*0.48*	*308.60*	*1122.28*	*1305.57*
3	*3.03*	*505.13*	*4.08*	*512.24*	*706.58*
4	*11.59*	*242.85*	*0.72*	*255.15*	*350.84*
5 (pool lid)	*33.90*	*1373.16*	*2.45*	*1409.51*†††	*1418.98*
ONE METER FROM THE 100-TON HI-TRAC					
1	*104.52*	*47.24*	*43.52*	*195.28*	*218.65*
2	*357.48*	*4.37*	*101.86*	*463.71*	*545.32*
3	*43.02*	*82.69*	*18.06*	*143.78*	*186.79*

Note: MPC internal water level is 10 inches below the MPC lid.

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

††† Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.3

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~35,000~~*42,500* MWD/MTU AND 5-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	***Totals with BPRAs (mrem/hr)***
ADJACENT TO THE 100-TON HI-TRAC					
1	*9.85*	*207.47*	*2.55*	*219.87*	*221.81*
2	*472.26*	*0.32*	*19.09*	*491.68*	*595.39*
3	*2.44*	*505.06*	*0.65*	*508.15*	*702.32*
4	*11.58*	*242.84*	*0.65*	*255.07*	*350.75*
5 (pool lid)	*33.76*	*1372.10*	*2.42*	*1408.27*†††	*1417.69*
ONE METER FROM THE 100-TON HI-TRAC					
1	*60.03*	*31.57*	*2.50*	*94.10*	*107.16*
2	*205.72*	*2.47*	*7.02*	*215.21*	*260.67*
3	*24.25*	*58.67*	*0.81*	*83.73*	*112.41*

Note: MPC internal water level is 10 inches below the MPC lid.

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

††† Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.4

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
52,500 MWD/MTU AND 10-YEAR COOLING

Dose Point Location	*Fuel Gammas (mrem/hr)*	*(n,γ) Gammas (mrem/hr)*	*^{60}Co Gammas (mrem/hr)*	*Neutrons (mrem/hr)*	*Totals (mrem/hr)*	*Totals with BPRAs (mrem/hr)*
ADJACENT TO THE 100-TON HI-TRAC						
1	15.08	19.66	363.97	267.22	665.92	673.90
2	422.99	82.07	0.54	137.11	642.70	868.38
3	6.44	3.07	317.36	333.60	660.47	877.80
3 (temp)	2.56	6.82	110.64	3.40	123.41	199.38
4	9.98	1.51	161.68	280.56	453.72	569.32
4 (outer)	2.72	0.94	40.25	189.66	233.58	262.67
5 (pool lid)	67.77	27.24	1835.58	1855.60	3786.19	3846.35
5 (transfer)	129.15	0.97	2049.81	1036.11	3216.04	3279.10
5(t-outer)	32.69	0.52	219.91	413.26	666.39	682.49
ONE METER FROM THE 100-TON HI-TRAC						
1	54.58	11.10	53.05	43.47	162.20	191.31
2	180.35	24.93	5.95	51.68	262.90	360.88
3	23.14	6.03	48.23	21.49	98.89	146.64
3 (temp)	23.01	6.38	40.21	7.48	77.08	119.42
4	3.39	0.26	49.91	69.71	123.27	159.08
5 (transfer)	50.98	0.21	872.89	288.98	1213.06	1237.44
5(t-outer)	7.63	0.94	77.61	83.06	169.24	172.14

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 3(temp) represents dose location 3 with temporary shielding installed.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.5

DOSE RATES FROM THE 125-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
42,500 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	***Fuel Gammas (mrem/hr)***	***(n,γ) Gammas (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***	***Totals with BPRAs (mrem/hr)***
ADJACENT TO THE 125-TON HI-TRAC						
1	*7.52*	*14.41*	*103.35*	*84.68*	*209.96*	*211.09*
2	*67.09*	*41.11*	*0.01*	*50.69*	*158.90*	*173.04*
3	*0.74*	*1.72*	*37.27*	*140.76*	*180.48*	*195.22*
4	*24.90*	*1.76*	*253.74*	*160.94*	*441.34*	*548.51*
4 (outer)	*3.01*	*1.26*	*31.51*	*3.37*	*39.15*	*52.33*
5 (pool)	*35.11*	*0.65*	*390.89*	*556.82*	*983.46*	*990.61*
5 (transfer)	*29.88*	*0.87*	*310.36*	*89.13*	*430.24*	*434.47*
ONE METER FROM THE 125-TON HI-TRAC						
1	*9.03*	*5.22*	*11.30*	*13.49*	*39.04*	*40.88*
2	*29.65*	*12.87*	*0.36*	*18.28*	*61.16*	*67.28*
3	*3.40*	*3.08*	*7.45*	*12.29*	*26.23*	*30.06*
4	*6.93*	*0.42*	*61.09*	*16.53*	*84.98*	*110.66*
5 (transfer)	*14.13*	*0.17*	*156.96*	*15.93*	*187.18*	*189.93*

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.6

ANNUAL DOSE AT 200 METERS FROM A SINGLE HI-STORM OVERPACK WITH AN MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL†

Dose Component	***52,500*** **MWD/MTU 5-Year Cooling (mrem/yr)**
Fuel gammas††	*16.52*
^{60}Co Gammas	*2.17*
Neutrons	*1.50*
Total	*20.19*

† 8760 hour annual occupancy is assumed.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS ISFSI CONFIGURATIONS
~~45,000~~*52,500* MWD/MTU AND 5-YEAR COOLING ZIRCALOY CLAD FUEL†

Distance	**A Side of Overpack (mrem/yr)**	**B Top of Overpack (mrem/yr)**	**C Side of Shielded Overpack (mrem/yr)**
100 meters	*129.0*	*1.59*	*25.80*
150 meters	*45.6*	*0.61*	*9.12*
200 meters	*19.9*	*0.27*	*3.98*
250 meters	*9.72*	*0.13*	*1.94*
300 meters	*5.20*	*6.57e-2*	*1.04*
350 meters	*3.05*	*3.35e-2*	*0.61*
400 meters	*1.75*	*1.77e-2*	*0.35*

† 8760 hour annual occupancy is assumed.

Table 5.4.8

DOSE RATES AT THE CENTERLINE OF THE OVERPACK FOR DESIGN BASIS STAINLESS STEEL CLAD FUEL *WITHOUT BPRAs*

Dose Point[†] Location	**Fuel Gammas[††] (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
MPC-24 (40,000 MWD/MTU AND 8-YEAR COOLING)				
2 (Adjacent)	36.97	0.02	1.11	38.10
2 (One Meter)	18.76	0.17	0.50	19.43
MPC-32 (40,000 MWD/MTU AND 9-YEAR COOLING)				
2 (Adjacent)	*37.58*	*0.00*	*1.49*	*39.08*
2 (One Meter)	*18.74*	*0.25*	*0.58*	*19.57*
MPC-68 (22,500 MWD/MTU AND 10-YEAR COOLING)				
2 (Adjacent)	17.79	0.01	0.10	17.90
2 (One Meter)	8.98	0.13	0.04	9.15

† Refer to Figure 5.1.1.

†† Gammas generated by neutron capture are included with fuel gammas.

Table 5.4.9

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~30,000~~*40,000* MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*37.93*	*13.18*	*886.50*	*175.78*	*1113.40*
2	*874.39*	*67.70*	*0.50*	*102.55*	*1045.14*
3	*3.61*	*1.36*	*841.96*	*107.82*	*954.74*
3 (temp)	*1.81*	*2.40*	*269.98*	*0.95*	*275.14*
4	*4.91*	*0.58*	*210.60*	*96.65*	*312.73*
4 (outer)	*1.39*	*0.39*	*55.78*	*57.80*	*115.36*
5 (pool lid)	*115.98*	*15.68*	*3959.26*	*1038.03*	*5128.95*
5(transfer lid)	*133.91*	*0.75*	*4483.29*	*642.74*	*5260.69*
5 (t-outer)	*48.17*	*0.30*	*410.54*	*239.91*	*698.93*
ONE METER FROM THE 100-TON HI-TRAC					
1	*117.75*	*8.54*	*82.10*	*31.38*	*239.76*
2	*370.80*	*19.10*	*6.92*	*37.88*	*434.70*
3	*29.23*	*3.30*	*132.29*	*8.14*	*172.96*
3 (temp)	*29.15*	*3.40*	*103.55*	*3.75*	*139.85*
4	*1.98*	*0.11*	*70.28*	*20.16*	*92.52*
5(transfer lid)	*65.14*	*0.54*	*1996.78*	*176.55*	*2239.00*
5 (t-outer)	*8.27*	*0.60*	*168.35*	*49.87*	*227.10*

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 3(temp) represents dose location 3 with temporary shielding installed.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.10

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~45,000~~*50,000* MWD/MTU AND ~~12~~*10*-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*14.98*	*19.48*	*502.66*	*259.77*	*796.88*
2	*352.56*	*101.89*	*0.32*	*162.51*	*617.28*
3	*1.35*	*2.01*	*477.40*	*159.33*	*640.09*
3 (temp)	*0.67*	*3.54*	*153.08*	*1.41*	*158.70*
4	*1.68*	*0.85*	*119.41*	*142.82*	*264.76*
4 (outer)	*0.49*	*0.57*	*31.63*	*85.42*	*118.11*
5 (pool lid)	*45.87*	*23.17*	*2244.94*	*1533.96*	*3847.94*
5(transfer lid)	*57.01*	*1.11*	*2542.07*	*949.82*	*3550.01*
5 (t-outer)	*18.73*	*0.45*	*232.78*	*354.53*	*606.49*
ONE METER FROM THE 100-TON HI-TRAC					
1	*48.56*	*12.62*	*46.55*	*46.37*	*154.10*
2	*152.83*	*28.64*	*3.34*	*56.28*	*241.09*
3	*12.03*	*4.87*	*75.01*	*12.03*	*103.95*
3 (temp)	*12.00*	*5.03*	*58.71*	*5.54*	*81.29*
4	*0.76*	*0.16*	*39.85*	*29.79*	*70.57*
5(transfer lid)	*26.03*	*0.80*	*1132.20*	*260.90*	*1419.92*
5 (t-outer)	*3.29*	*0.89*	*95.46*	*73.70*	*173.33*

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 3(temp) represents dose location 3 with temporary shielding installed.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.11

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
32,500 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	34.50	6.30	710.92	90.10	841.83	851.59
2	875.41	30.29	0.67	45.87	952.24	1212.20
3	18.05	1.06	732.97	104.16	856.23	1196.04
4	28.44	0.80	337.51	90.10	456.85	623.02
4 (outer)	7.59	0.32	84.67	61.27	153.84	195.62
5 (pool)	216.26	8.54	3950.57	614.70	4790.07	4866.44
5 (transfer)	305.05	0.35	4476.67	342.91	5124.97	5201.86
5(t-outer)	55.49	0.16	381.36	130.67	567.68	581.74
ONE METER FROM THE 100-TON HI-TRAC						
1	115.62	3.92	103.64	14.45	237.62	271.55
2	382.92	9.12	7.75	17.78	417.58	532.82
3	50.27	2.22	100.77	6.74	160.00	225.40
4	8.54	0.12	100.64	22.21	131.51	180.87
5 (transfer)	128.73	0.17	1811.31	95.10	2035.31	2067.33
5(t-outer)	12.84	0.35	156.29	26.61	196.09	199.29

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.12

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
45,000 MWD/MTU AND 10-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	15.12	13.36	441.43	190.84	660.75	670.51
2	405.59	64.16	0.42	97.15	567.32	827.28
3	7.71	2.24	455.12	220.62	685.68	1025.49
4	12.19	1.70	209.57	190.82	414.27	580.44
4 (outer)	3.10	0.67	52.57	129.78	186.12	227.90
5 (pool)	102.13	18.08	2453.00	1301.93	3875.14	3951.51
5 (transfer)	142.98	0.73	2779.67	726.34	3649.72	3726.61
5(t-outer)	24.56	0.35	236.79	276.77	538.48	552.54
ONE METER FROM THE 100-TON HI-TRAC						
1	52.82	8.30	64.35	30.60	156.07	190.00
2	173.93	19.20	6.94	38.22	238.29	353.53
3	23.08	4.70	62.57	14.28	104.64	170.04
4	3.39	0.24	62.49	47.05	113.18	162.54
5 (transfer)	58.41	0.36	1124.68	201.44	1384.89	1416.91
5(t-outer)	5.66	0.75	97.04	56.37	159.82	163.02

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.13

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS WITH FOUR DAMAGED FUEL CONTAINERS MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL 42,500 MWD/MTU AND 5-YEAR COOLING WITHOUT BPRAs

Dose Point† Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	48.60	15.24	626.44	227.63	917.91
2	975.75	55.99	0.92	90.86	1123.53
3	18.96	2.89	546.22	323.32	891.38
ONE METER FROM THE 100-TON HI-TRAC					
1	135.96	8.42	91.31	34.38	270.06
2	425.52	18.08	7.07	35.33	486.01
3	61.73	4.80	83.00	19.58	169.11

† Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.14

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS WITH SIXTEEN DAMAGED FUEL CONTAINERS MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL 40,000 MWD/MTU AND 5-YEAR COOLING

Dose Point[†] Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	89.97	18.53	886.50	327.43	1322.43
2	819.00	69.93	0.57	105.17	994.66
3	4.40	2.56	841.96	236.47	1085.38
ONE METER FROM THE 100-TON HI-TRAC					
1	137.42	9.87	82.10	44.92	274.30
2	358.93	21.26	5.87	40.31	426.38
3	40.53	4.65	132.29	16.75	194.21

† Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.15

DOSE RATES DUE TO BPRAs AND TPDs FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS

	MPC-24		MPC-32	
Dose Point Location	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC				
1	7.98	0.00	9.76	0.01
2	225.68	0.02	259.96	0.03
3	217.33	197.20	339.81	304.82
3 (temp)	75.97	68.99	117.84	106.24
4	115.60	106.71	166.17	156.16
4 (outer)	29.09	27.12	41.78	39.32
5 (pool lid)	60.16	0.01	76.37	0.00
5(transfer lid)	63.06	0.00	76.89	0.00
5(t-outer)	16.10	0.00	14.06	0.00
ONE METER FROM THE 100-TON HI-TRAC				
1	29.11	0.18	33.93	0.24
2	97.98	0.77	115.24	1.04
3	47.75	40.55	65.40	57.95
3 (temp)	42.34	35.76	56.45	50.01
4	35.81	33.37	49.36	47.19
5(transfer lid)	24.38	0.00	32.02	0.00
5(t-outer)	2.90	0.00	3.20	0.00

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 3(temp) represents dose location 3 with temporary shielding installed.*
- *Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.16

DOSE RATES DUE TO CRAs FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS

	MPC-24		MPC-32	
Dose Point Location	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC				
1	5.39	1.02	3.25	0.68
2	0.02	0.00	0.01	0.00
3	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5 (pool lid)	919.75	170.85	1141.11	213.21
5(transfer lid)	1116.24	212.79	1473.52	279.02
5(t-outer)	1.00	0.17	0.68	0.13
ONE METER FROM THE 100-TON HI-TRAC				
1	1.18	0.20	0.69	0.13
2	0.14	0.02	0.03	0.01
3	0.01	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5(transfer lid)	169.44	32.19	192.55	36.32
5(t-outer)	6.22	1.16	6.64	1.29

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.17

DOSE RATES DUE TO APSRs FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS

	MPC-24			MPC-32		
Dose Point Location	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**	**Config. 3 (mrem/hr)**	**Config. 1 (mrem/hr)**	**Config. 2 (mrem/hr)**	**Config. 3 (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC						
1	12.43	2.34	12.42	7.51	1.58	7.67
2	0.03	0.00	7.59	0.01	0.00	0.17
3	0.00	0.00	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00	0.00	0.00
5 (pool lid)	1996.3	371.98	1940.91	2414.85	453.81	2687.17
5(transfer)	2294.93	435.67	2285.99	3021.44	570.37	2855.84
5(t-outer)	2.24	0.38	2.51	1.52	0.30	1.62
ONE METER FROM THE 100-TON HI-TRAC						
1	2.69	0.46	3.45	1.57	0.31	1.66
2	0.32	0.04	2.71	0.07	0.01	0.12
3	0.02	0.00	0.04	0.01	0.01	0.00
4	0.00	0.00	0.00	0.00	0.00	0.00
5(transfer)	359.71	67.81	356.54	406.10	76.30	396.70
5(t-outer)	13.27	2.47	13.42	14.18	2.69	14.33

Notes:

- *Refer to Figures 5.1.2 and 5.1.4 for dose locations.*
- *Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.*
- *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.4.18

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL

Assembly	***Active fuel length (inch)***	***Neutrons per sec per inch***	***Neutrons per sec per inch with Sb-Be source***	***Reference for neutrons per sec per inch***
7x7 design basis	*144*	*9.17E+5*	*N/A*	*Table 5.2.17 - 40 GWD/MTU and 5 year cooling*
6x6 design basis	*110*	*2.0E+5*	*2.6E+5*	*Table 5.2.18*
6x6 design basis MOX	*110*	*3.06E+5*	*3.66E+5*	*Table 5.2.23*

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CHAPTER 6†: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel in accordance with 10CFR72.124. The results of this evaluation demonstrate that the HI-STORM 100 System is in full compliance with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, and thus, fulfills the following acceptance criteria:

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both. ~~Where solid neutron absorbing materials are used, the design should provide for a positive means to verify their continued efficacy during the storage period.~~

4. Criticality safety of the cask system should not rely on use of the following credits:

   a. burnup of the fuel
   b. fuel-related burnable neutron absorbers
   c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance test.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STORM 100 System design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. Analyses for the HI-STAR 100 System, which are applicable to the HI-STORM 100 System, have been previously submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

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† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter 1,* Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1], 10CFR72.124 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, these results demonstrate that the HI-STORM 100 System is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM 100 System when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM 100 System depends on the following ~~three~~*four* principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24, *MPC-24E and MPC-24EF*);

2. The incorporation of permanent fixed neutron-absorbing panels (Boral) in the fuel basket structure;~~ and~~

3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel~~.~~*; and*

4. *An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel with higher enrichments in the MPC-24, MPC-24E and MPC-24EF, and for loading/unloading fuel in the MPC-32.*

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design parameters important to criticality safety, and thus, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM 100 System is designed such that the fixed neutron absorber (Boral) will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. Therefore, in accordance with ~~NUREG-1536~~*10CFR72.124(b)*, there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber~~, as required by 10CFR72.124(b)~~.

Criticality safety of the HI-STORM 100 System does not rely on the use of any of the following credits:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 75 percent of the B-10 content for the fixed neutron absorber (Boral).

The following ~~two~~*four* interchangeable basket designs are available for use in the HI-STORM 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment *and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,*

- *a 24-cell basket (MPC-24E) for intact and damaged PWR fuel assemblies. This is a variation of the MPC-24, with an optimized cell arrangement, increased ^{10}B content in the Boral and with four cells capable of accommodating either intact fuel or a damaged fuel container (DFC). Additionally, a variation in the MPC-24E, designated MPC-24EF, is designed for intact and damaged PWR fuel assemblies and PWR fuel debris. The MPC-24E and MPC-24EF is designed for fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,*

- *a 32-cell basket (MPC-32), designed for intact PWR fuel assemblies of a specified maximum enrichment and minimum soluble boron concentration for loading/unloading, and*

- a 68-cell basket (MPC-68), designed for both intact and damaged BWR fuel assemblies with a specified maximum planar-average enrichment. Additionally,~~-a~~ variation*s* in the MPC-68, designated MPC-68F *and MPC-68FF,* ~~is~~*are* designed for *intact and damaged BWR fuel assemblies and* BWR fuel debris with a specified maximum planar-average enrichment.

The HI-STORM 100 System includes the HI-TRAC transfer cask and the HI-STORM storage cask. The HI-TRAC transfer cask is required for loading and unloading fuel into the MPC and for transfer of the MPC into the HI-STORM storage cask. HI-TRAC uses a lead shield for gamma radiation and a water-filled jacket for neutron shielding. The HI-STORM storage cask uses concrete as a shield for both gamma and neutron radiation. Both the HI-TRAC transfer cask and the HI-STORM storage cask, as well as the HI-STAR System[†], accommodate the interchangeable MPC designs. The three cask designs (HI-STAR, HI-STORM, and HI-TRAC) differ only in the overpack reflector materials (steel for HI-STAR, concrete for HI-STORM, and lead for HI-TRAC), which do not significantly affect the reactivity. Consequently, analyses for the HI-STAR[‡] System ~~(e.g., determination of bounding assembly class dimensions, determination of worst case combination of manufacturing tolerances, evaluation of the reactivity effect of various conditions of moderation, and the evaluation of damaged fuel) are applicable to the HI-STORM 100 System. Therefore, the analyses discussed in Sections 6.2, 6.3, and 6.4 of this chapter were taken directly from the HI-STAR applications (USNRC Docket Numbers 72-1008 and 71-9261).~~ *are directly applicable to the HI-STORM 100 system and vice versa. Therefore, the majority of criticality calculations to support both the HI-STAR and the HI-STORM System have been performed for only one of the two systems, namely the HI-STAR System. Only a selected number of analyses has been performed for both systems to demonstrate that this approach is valid. Therefore, unless specifically noted otherwise, all analyses documented throughout this chapter have been performed for the HI-STAR System. For the cases where analyses were performed for both the HI-STORM and HI-STAR System, this is clearly indicated.*

The HI-STORM 100 System for storage (concrete overpack) is dry (no moderator), and thus, the reactivity is very low (k_{eff} <0.4052). However, the HI-STORM 100 System for cask transfer (HI-TRAC, lead overpack) is flooded for loading and unloading operations, and thus, represents the limiting case in terms of reactivity. ~~Soluble boron credit is not required for loading/unloading operations.~~

The MPC-24EF contains the same basket as the MPC-24E. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-24E and MPC-24EF. Therefore, all criticality results obtained for the MPC-24E are valid for the MPC-24EF and no separate analyses for the MPC-24EF are necessary.

The MPC-68FF contains the same basket as the MPC-68. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-68 are valid for the MPC-68FF and no separate analyses for the MPC-68FF are necessary.

Confirmation of the criticality safety of the HI-STORM 100 System was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations were made with NITAWL-KENO5a from the SCALE-4.3 package [6.4.1]. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-

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† Analyses for the HI-STAR System have previously been submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

To assess the incremental reactivity effects due to manufacturing tolerances, CASMO-3, a two-dimensional transport theory code [6.1.9-6.1.12] for fuel assemblies, *and MCNP4a [6.1.4] were*~~as~~ used~~ to assess the incremental reactivity effects due to manufacturing tolerances~~. The CASMO-3 *and MCNP4a* calculations identify those tolerances that cause a positive reactivity effect, enabling the *subsequent* Monte Carlo code input to define the worst case (most conservative) conditions. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a and KENO5a) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24, *MPC-24E and MPC-24EF*) or cell spacing (*MPC-32 and* MPC-68),~~ and~~ (3) the ^{10}B loading of the neutron absorber panels, *and (4) the soluble boron concentration in the water*. The critical experiment benchmarking~~, which is taken from the HI-STAR applications (Docket Numbers 72-1008 and 71-9261),~~ is presented in Appendix 6.A.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington D.C., January 1997.

- 10CFR72.124, Criteria For Nuclear Criticality Safety.

- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- In accordance with NUREG-1536, no credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.

- In accordance with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber.

- The fuel stack density is conservatively assumed to be 96% of theoretical (10.522 g/cm^3) for all criticality analyses. Fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels may be slightly greater than 96% of theoretical, the actual stack density will be less.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be ~~pure, unborated~~ water, *with or without soluble boron,* at a temperature *and density* corresponding to the highest reactivity within the expected operating range~~ (i.e., water density of 1.000 g/cc)~~.

- *When credit is taken for soluble boron, a ^{10}B content of 18.0 wt% in boron is assumed.*

- Neutron absorption in minor structural members and *optional* heat conduction elements is neglected, i.e., spacer grids, basket supports, and *optional* aluminum heat conduction elements are replaced by water.

- In compliance with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded *with pure unborated water*.

- Planar-averaged enrichments are assumed for BWR fuel. (In accordance with NUREG-1536, analysis is presented in Appendix 6.B† to demonstrate that the use of planar-average enrichments produces conservative results.)

- In accordance with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- For evaluation of the bias, all benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000, in accordance with NUREG-1536.

† ~~Taken directly from the HI-STAR applications (Docket Numbers 72-1008 and 71-9261).~~

- *The water reflector above and below the fuel is assumed to be unborated water, even if borated water is used in the fuel region.*

- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.

- For intact fuel assemblies, as defined in ~~Chapter 12~~*the Certificate of Compliance*, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

~~Because the HI-STAR and HI-STORM Systems differ only in the overpack material, the limiting cases are assumed to be the same for both systems. Consequently, to bound all fuel types and basket configurations that were previously demonstrated to be acceptable in the HI-STAR System, explicit calculations were performed for each of the limiting cases in the HI-STORM System.~~

Results of the design basis criticality safety calculations for single internally flooded HI-TRAC transfer casks with full water reflection on all sides (limiting cases for the HI-STORM 100 System) *loaded with intact fuel assemblies* are listed in Tables 6.1.1 through 6.1.*8*~~3~~, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. To demonstrate that the overpack material does not significantly affect the reactivity, results of the design basis criticality safety calculations for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System) are listed in Tables 6.1.1 through 6.1.*8*~~3~~ for comparison. In addition, a few results for single internally dry (no moderator) HI-STORM storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are listed to confirm the low reactivity of the HI-STORM 100 System in storage.

For each of the MPC designs, *minimum soluble boron concentration (if applicable)* and fuel assembly classes[††], Tables 6.1.1 through 6.1.~~3~~*8* list the bounding maximum k_{eff} value, and the associated maximum allowable enrichment. The maximum allowed enrichments *and the minimum soluble boron concentrations* are defined in ~~the Technical Specifications contained in Chapter 12~~ *Appendix B to the Certificate of Compliance.* ~~Maximum k_{eff} values for each of t~~*T*he

[††] For each array size (e.g., 6x6, 7x7, 14x14, etc.), the fuel assemblies have been subdivided into a number of assembly classes, where an assembly class is defined in terms of the (1) number of fuel rods; (2) pitch; (3) number and location of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Section 6.2.

candidate fuel assemblies~~and basket configurations~~, that are bounded by those listed in Tables 6.1.1 through 6.1.*8*~~3~~, are given in Section 6.2.

Results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) loaded with damaged fuel assemblies or a combination of intact and damaged fuel assemblies are listed in Tables 6.1.9 through 6.1.11. The results include the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs qualified for damaged fuel and/or fuel debris (MPC-24E, MPC-24EF, MPC-68, MPC-68F and MPC-68FF), Tables 6.1.9 through 6.1.11 indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum k_{eff} value and the associated maximum allowable enrichment. For the permissible location of DFCs see Subsection 6.4.4.2. The maximum allowed enrichments are defined in Appendix B to the Certificate of Compliance.

A table listing the maximum k_{eff} (including bias, uncertainties, and calculational statistics), calculated k_{eff}, standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations is provided in Appendix 6.C. These results confirm that the maximum k_{eff} values for the HI-STORM 100 System are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STORM 100 System to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM 100 System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

Table 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24
(no soluble boron)

Fuel Assembly Class	**Maximum Allowable Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**		
		HI-STORM	**HI-TRAC**	**HI-STAR**
14x14A	4.6	0.2938	0.9365	0.9383
14x14B	4.6	---	0.9313	0.9323
14x14C	4.6	---	0.9~~360~~*395*	0.9~~361~~*400*
14x14D	4.0	---	0.8583	0.8576
14x14E	*5.0*	---	*0.7702*	*0.7715*
15x15A	4.1	---	0.9292	0.9301
15x15B	4.1	---	0.9467	0.9473
15x15C	4.1	---	0.9448	0.9444
15x15D	4.1	---	0.9447	0.9440
15x15E	4.1	---	0.9474	0.9475
15x15F	4.1	0.3416	0.9468††	0.9478†††
15x15G	4.0	---	0.8972	0.8986
15x15H	*3.8*	---	*0.9411*	*0.9411*
16x16A	4.6	0.3273	0.9363	0.9383
17x17A	4.0	0.3082	0.9433	0.9452
17x17B	4.0	---	0.9412	0.9436
17x17C	4.0	---	0.9421	0.9427

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9471.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9466.

Table 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24 WITH 400 PPM SOLUBLE BORON

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Maximum† k_{eff}***		
		HI-STORM	***HI-TRAC***	***HI-STAR***
14x14A	*5.0*	---	---	*0.8986*
14x14B	*5.0*	---	---	*0.8977*
14x14C	*5.0*	---	---	*0.9042*
14x14D	*5.0*	---	---	*0.8627*
14x14E	*5.0*	---	---	*0.7176*
15x15A	*5.0*	---	---	*0.9209*
15x15B	*5.0*	---	---	*0.9362*
15x15C	*5.0*	---	---	*0.9351*
15x15D	*5.0*	---	---	*0.9352*
15x15E	*5.0*	---	---	*0.9388*
15x15F	*5.0*	*0.4111*	*0.9410*	*0.9402*
15x15G	*5.0*	---	---	*0.9022*
15x15H	*5.0*	---	*0.9447*	*0.9447*
16x16A	*5.0*	---	---	*0.9058*
17x17A	*5.0*	---	---	*0.9371*
17x17B	*5.0*	---	---	*0.9372*
17x17C	*5.0*	---	*0.9385*	*0.9386*

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E AND MPC-24EF (no soluble boron)

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Maximum† k_{eff}***		
		HI-STORM	***HI-TRAC***	***HI-STAR***
14x14A	*5.0*	---	---	*0.9380*
14x14B	*5.0*	---	---	*0.9312*
14x14C	*5.0*	---	---	*0.9356*
14x14D	*5.0*	---	---	*0.8875*
14x14E	*5.0*	---	---	*0.7651*
15x15A	*4.5*	---	---	*0.9336*
15x15B	*4.5*	---	---	*0.9465*
15x15C	*4.5*	---	---	*0.9462*
15x15D	*4.5*	---	---	*0.9440*
15x15E	*4.5*	---	---	*0.9455*
15x15F	*4.5*	*0.3699*	*0.9465*	*0.9468*
15x15G	*4.5*	---	---	*0.9054*
15x15H	*4.2*	---	---	*0.9423*
16x16A	*5.0*	---	---	*0.9341*
17x17A	*4.4*	---	*0.9467*	*0.9447*
17x17B	*4.4*	---	---	*0.9421*
17x17C	*4.4*	---	---	*0.9433*

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E AND MPC-24EF WITH 300 PPM SOLUBLE BORON

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Maximum† k_{eff}***		
		HI-STORM	***HI-TRAC***	***HI-STAR***
14x14A	*5.0*	---	---	*0.8963*
14x14B	*5.0*	---	---	*0.8974*
14x14C	*5.0*	---	---	*0.9031*
14x14D	*5.0*	---	---	*0.8588*
14x14E	*5.0*	---	---	*0.7249*
15x15A	*5.0*	---	---	*0.9161*
15x15B	*5.0*	---	---	*0.9321*
15x15C	*5.0*	---	---	*0.9271*
15x15D	*5.0*	---	---	*0.9290*
15x15E	*5.0*	---	---	*0.9309*
15x15F	*5.0*	*0.3897*	*0.9333*	*0.9332*
15x15G	*5.0*	---	---	*0.8972*
15x15H	*5.0*	---	*0.9399*	*0.9399*
16x16A	*5.0*	---	---	*0.9021*
17x17A	*5.0*	---	*0.9320*	*0.9332*
17x17B	*5.0*	---	---	*0.9316*
17x17C	*5.0*	---	---	*0.9312*

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 WITH 1900 PPM SOLUBLE BORON

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Maximum† k_{eff}***		
		HI-STORM	***HI-TRAC***	***HI-STAR***
14x14A	*4.1*	---	---	*0.8372*
14x14B	*4.1*	---	---	*0.8626*
14x14C	*4.1*	---	---	*0.8776*
14x14D	*4.1*	---	---	*0.8405*
14x14E	*4.1*	---	---	*0.6288*
15x15A	*4.1*	---	---	*0.9075*
15x15B	*4.1*	---	---	*0.9239*
15x15C	*4.1*	---	---	*0.9108*
15x15D	*4.1*	---	---	*0.9375*
15x15E	*4.1*	---	---	*0.9348*
15x15F	*4.1*	*0.4691*	*0.9403*	*0.9411*
15x15G	*4.1*	---	---	*0.8980*
15x15H	*4.1*	---	---	*0.9267*
16x16A	*4.1*	---	---	*0.8831*
17x17A	*4.1*	---	---	*0.9105*
17x17B	*4.1*	---	---	*0.9309*
17x17C	*4.1*	---	*0.9365*	*0.9355*

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.6

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 WITH 2600 PPM SOLUBLE BORON

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***	***Maximum† k_{eff}***		
		HI-STORM	***HI-TRAC***	***HI-STAR***
14x14A	*5.0*	---	---	*0.8362*
14x14B	*5.0*	---	---	*0.8633*
14x14C	*5.0*	---	---	*0.8901*
14x14D	*5.0*	---	---	*0.8485*
14x14E	*5.0*	---	---	*0.6240*
15x15A	*5.0*	---	---	*0.9121*
15x15B	*5.0*	---	---	*0.9313*
15x15C	*5.0*	---	---	*0.9181*
15x15D	*5.0*	---	---	*0.9466*
15x15E	*5.0*	---	---	*0.9434*
15x15F	*5.0*	*0.5142*	*0.9470*	*0.9483*
15x15G	*5.0*	---	---	*0.9135*
15x15H	*5.0*	---	---	*0.9317*
16x16A	*5.0*	---	---	*0.8924*
17x17A	*5.0*	---	---	*0.9160*
17x17B	*5.0*	---	---	*0.9371*
17x17C	*5.0*	---	*0.9436*	*0.9437*

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.*7*~~2~~

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 *AND MPC-68FF*

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	**HI-TRAC**	**HI-STAR**
6x6A	2.7††	---	0.7*886*~~599~~	0.7*888*~~602~~†††
6x6B‡	2.7††	---	0.7*833*~~625~~	0.7*824*~~611~~†††
6x6C	2.7††	0.2759	0.8024	0.8021†††
7x7A	2.7††	---	0.79*63*~~56~~	0.797*4*~~3~~†††
7x7B	4.2	0.*4061*~~3826~~	0.938*5*~~0~~	0.93*86*~~78~~
8x8A	2.7††	---	0.76*90*~~62~~	0.76*97*~~85~~†††
8x8B	4.2	*0.3934*~~—~~	0.9*427*~~375~~	0.9*416*~~368~~
8x8C	4.2	0.3714	0.9402	0.9425
8x8D	4.2	---	0.9*408*~~360~~	0.9*403*~~366~~
8x8E	4.2	---	0.9309	0.9312
8x8F	*4.0*	---	*0.9396*	*0.9411*

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† This calculation was performed for 3.0% planar-average enrichment, however, the actual fuel and ~~Technical Specifications~~ *Certificate of Compliance* are limited to maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} value is conservative.

††† This calculation was performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68 is 0.0372 g/cm^2. Therefore, the listed maximum k_{eff} value is conservative.

‡ Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the ~~Technical Specifications, Chapter 12~~*Certificate of Compliance.*

Table 6.1.*7*~~2~~ (continued)

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 *AND MPC-68FF*

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
9x9A	4.2	0.3365	0.9434	0.9417
9x9B	4.2	---	0.94*17*~~16~~	0.9*436*~~388~~
9x9C	4.2	---	0.9377	0.9395
9x9D	4.2	---	0.9387~~1~~	0.9394~~2~~
9x9E	4.~~2~~*0*	—	0.94~~03~~*02*	0.94~~06~~*01*
9x9F	4.~~2~~*0*	---	0.9~~366~~*402*	0.9~~377~~*401*
9x9G	*4.2*	---	*0.9307*	*0.9309*
10x10A	4.2	0.3379	0.9448‡‡	0.9457*
10x10B	4.2	---	0.9443	0.9436
10x10C	4.2	---	0.9~~002~~*430*	0.~~8990~~*9433*
10x10D	4.0	---	0.9383	0.9376
10x10E	4.0	---	0.9157	0.9185

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

‡‡ KENO5a verification calculation resulted in a maximum k_{eff} of 0.9451.

* KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

Table 6.1.*8*~~3~~

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**	**Maximum† k_{eff}**		
		HI-STORM	**HI-TRAC**	**HI-STAR**
6x6A	2.7††	---	0.7*886*~~599~~	0.7*888*~~602~~
6x6B†††	2.7	---	0.7*833*~~625~~	0.7*824*~~611~~
6x6C	2.7	0.2759	0.8024	0.8021
7x7A	2.7	---	0.79*63*~~56~~	0.797*4*~~3~~
8x8A	2.7	---	0.76*90*~~62~~	0.7*697*~~685~~

Notes:

1. The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

2. These calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

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† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† These calculations were performed for 3.0% planar-average enrichment, however, the actual fuel and ~~Technical Specifications~~*Certificate of Compliance* are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} values are conservative.

††† Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is specified in the ~~Technical Specifications, Chapter 12~~*Certificate of Compliance.*

Table 6.1.9

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-24E AND MPC-24EF WITH UP TO 4 DFCs

Fuel Assembly Class	***Maximum Allowable Enrichment (wt% ^{235}U)***		***Maximum k_{eff}***	
	Intact Fuel	***Damaged Fuel***	***HI-TRAC***	***HI-STAR***
All PWR Classes	*4.0*	*4.0*	*0.9486*	*0.9480*

Table 6.1.10

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68, MPC-68F AND MPC-68FF WITH UP TO 68 DFCs

Fuel Assembly Class	***Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)***		***Maximum k_{eff}***	
	Intact Fuel	***Damaged Fuel***	***HI-TRAC***	***HI-STAR***
6x6A, 6x6B, 6x6C, 7x7A, 8x8A	*2.7*	*2.7*	*0.8024*	*0.8021*

Table 6.1.11

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68 AND MPC-68FF WITH UP TO 16 DFCs

Fuel Assembly Class	***Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)***		***Maximum k_{eff}***	
	Intact Fuel	***Damaged Fuel***	***HI-TRAC***	***HI-STAR***
All BWR Classes	*3.7*	*4.0*	*0.9328*	*0.9328*